TVA

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: **WATTS BAR NUCLEAR PLANT (WBN) UNIT 2 – REQUEST FOR ADDITIONAL INFORMATION REGARDING SEVERE ACCIDENT MANAGEMENT ALTERNATIVES (TAC NO. MD8203)**

References:

1. TVA letter dated January 27, 2009, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Supplemental Environmental Impact Statement - Severe Accident Management Alternatives (TAC MD8203)" (ML090360588)

2. NRC letter dated November 30, 2009, "Watts Bar Nuclear Plant, Unit 2 – Request for Additional Information Regarding Severe Accident Management Alternatives (TAC NO. MD8203)" (ML092230024)

The purpose of this letter is to provide responses to the NRC request for additional information regarding the Severe Accident Management Alternatives (SAMA) discussed in the WBN Unit 2 Final Environmental Impact Statement submitted to the NRC (Reference 1). Enclosure 1 provides the NRC requests for additional information (Reference 2) and TVA's responses.

The new commitments are shown in Enclosure 2.

If you have any questions, please contact me at (423) 365-2351.

Sincerely,

M. Bajestani

Masoud Bajestani
Watts Bar Unit 2 Vice President

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NRR

Enclosures:

1. Responses to NRC Request for Additional Information
2. List of Commitments

cc (Enclosures):

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1. NRC Request

Provide the following information regarding the Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternative (SAMA) analysis:

a. Provide a breakdown of the internal event core damage frequency (CDF) by initiating event (including internal floods) that equals the total internal events CDF reported in the submittal. Provide the contribution from station blackout and anticipated transient without scram events if not separately provided in this listing.

TVA Response

The breakdown of initiating events including internal flooding is provided below:

Initiator Results for Model: WBN4SAM5 Master Frequency File: SAMA-CET				
Initiator		**CDF**	**Percent of Total**	**Initiating Event Frequency**
Very Small LOCA (<3/4-inch diameter, non-isolable)	SLOCAV	5.65E-06	37%	5.7770E-003
Total Loss of Essential Raw Cooling Water (ERCW)	ERCWTL	4.46E-06	29%	2.8970E-004
Total Loss of Component Cooling Water (CCS)	CCSTL	9.58E-07	6.2%	6.8230E-005
Loss of CCS Train A	CCSA	5.93E-07	3.9%	5.1270E-003
Loss of ERCW Train B	ERCWB	5.64E-07	3.7%	1.7300E-003
Loss of Offsite Power	LOSP	4.53E-07	3.0%	4.7850E-002
Small LOCA - Non Isolable	SLOCAN	3.13E-07	2.0%	4.9640E-004
Excessive LOCA (reactor vessel failure)	ELOCA	2.64E-07	1.7%	2.6360E-007
Loss of 125V Vital Battery Board II	LVBB2	2.12E-07	1.4%	4.0300E-003
Loss of 125V Vital Battery Board I	LVBB1	1.71E-07	1.1%	4.0680E-003
Loss of Primary Flow	LRCP	1.46E-07	0.95%	6.0570E-002
Loss of 120V Vital AC Board 1-III	LDCAC	1.10E-07	0.71%	4.4220E-002

Initiator Results for Model: WBN4SAM5 Master Frequency File: SAMA-CET				
Initiator		**CDF**	**Percent of Total**	**Initiating Event Frequency**
Turbine Trip	TTIE	1.01E-07	0.66%	8.6530E-001
Loss of ERCW Train A	ERCWA	9.71E-08	0.63%	1.6650E-003
Medium LOCA	MLOCA	8.46E-08	0.55%	2.6520E-005
Loss of Condenser Vacuum	LOCV	7.98E-08	0.52%	1.4560E-001
Partial Loss of Main Feedwater	PLMFW	6.52E-08	0.42%	3.2220E-001
Steam Generator Tube Rupture	SGTR	6.45E-08	0.42%	6.4880E-003
Loss of 120V Vital AC Board 1-II	LDBAC	4.39E-08	0.29%	4.5180E-002
Total Loss of Main Feedwater	TLMFW	4.30E-08	0.28%	7.9520E-002
Loss of 120V Vital AC Board 1-I	LDAAC	4.16E-08	0.27%	4.4720E-002
Reactor Trip	RTIE	3.46E-08	0.23%	5.1230E-001
Excessive Feedwater	EXMFW	2.90E-08	0.19%	5.5850E-002
Large LOCA	LLOCA	2.86E-08	0.19%	2.5460E-006
Loss of 120V Vital AC Board 1-IV	LDDAC	1.54E-08	0.10%	4.4240E-002
ISLOCA - RHR suction path	XS	1.39E-08	0.09%	1.8000E-006
Inadvertent Closure of all MSIVs	IMSIV	1.17E-08	0.08%	1.5200E-002
Inadvertent Safety Injection	ISI	8.17E-09	0.05%	1.5330E-002
Inadvertent Closure of one MSIV	MSIV	7.05E-09	0.05%	5.1650E-002
Loss of Plant Air	LOPA	6.05E-09	0.04%	9.8100E-003
Small LOCA -Isolable	SLOCAI	4.51E-09	0.03%	9.1840E-004
Steam Line Break Outside Containment	SLBOC	3.82E-09	0.02%	5.3740E-003
Steam Line Break Inside Containment	SLBIC	3.31E-09	0.02%	4.4340E-004
Core Power Excursion	CPEX	2.43E-09	0.02%	1.9250E-002
Inadvertent Opening of Main Steam Relief Valves	MSVO	1.87E-09	0.01%	2.9940E-003

Initiator Results for Model: WBN4SAM5 Master Frequency File: SAMA-CET				
Initiator		**CDF**	**Percent of Total**	**Initiating Event Frequency**
ISLOCA - RHR injection path	XI	5.35E-10	0.00%	1.3000E-006
Flooding Initiators				
Turbine Building Flood	FLTB	5.46E-09	0.04%	8.9910E-003
ERCW Strainer Room Train B Flood	FLPH1B	2.41E-07	1.57%	7.3780E-004
ERCW Strainer Room Train A Flood	FLPH1A	4.32E-07	2.81%	4.8150E-004
RWST Drained to Auxiliary Building Flood	FLAB3R	2.62E-09	0.02%	7.6760E-004
CST Drained to Auxiliary Building Flood	FLAB3C	1.53E-10	0.00%	6.4440E-006
ERCW Flood in Auxiliary Building	FLAB2	1.11E-11	0.00%	3.0630E-006
Total CDF		**1.537E-05**		

Station blackout (scenarios with failure of 6.9kV shutdown boards 1A and 1B – top events AAL and BAL failed) contribute 2.2%, or approximately 3.38E-7 to total CDF of 1.537E-5.

Anticipated transient without scram (ATWS) contributes 4.06% of total CDF (see response to SAMA 131). Based on the reference CDF of 1.537E-5, this corresponds to an ATWS CDF of 6.24E-7.

b. <u>*NRC Request*</u>

Describe the evolution of the Watts Bar Nuclear Plant (WBN) Level 1 and Level 2 PRA from the original Individual Plant Examination (IPE) submittal to the version used for the SAMA analysis. For each version of the PRA, provide the date, the CDF and large early release frequency (LERF), if calculated, and a summary description of the major changes that resulted in the increase/decrease in CDF and LERF relative to the prior version.

<u>TVA Response</u>

- The WBN IPE was issued September 1992. The CDF was calculated to be 3.3E-4/reactor-year and the LERF value was calculated to be 1.5E-5/reactor-year. This document was WBN's initial response to Generic Letter 88-20 and was issued prior to WBN fuel load.

- WBN IPE, Revision 1, was issued April 1994 also prior to WBN fuel load, and the purpose of the revision was to bring the IPE up to date with the plant configuration at fuel load. The CDF and LERF were calculated at 8E-5/reactor-year and 7.8E-6 respectively. Major updates included in this revision were:

    - A revision to the success criteria of the CCS to indicate that only one of the two train A pumps are required to support plant operation during normal conditions.
    - Provision for the use of nitrogen bottles for steam generator (SG) power-operated relief valves (PORVs) and auxiliary feedwater (AFW) flow control valves under station black (SBO) conditions.
    - Upgrade of human actions as the result of continued operator training and the plant procedures upgrade program including improvements made to the abnormal operating procedure for loss of CCS cooling.

- WBN PSA, Revision 2, was issued October 1997. This update consisted of a revision of the Level 1 probabilistic safety assessment (PSA) which incorporated changes made to the plant design as a result of the Unit 1 Severe Accident Mitigation Design Alternatives (SAMDA). The Level 2 model and containment release frequencies were not updated during this revision. Revision 2 estimates an average annual CDF of 4.4E-5, which is a decrease of 3.6E-05 from the average annual CDF reported in Revision 1. The principal drivers behind this reduction involved changes to the model in the areas of:
    - recovery from a loss of offsite power
    - high pressure recirculation
    - the CCS

  The first two items were identified through SAMDA. The offsite power recovery credited the use of the Unit 2 equipment to cross tie to the 500kV grid to the 161kV. Included in this category are changes to the models for the emergency diesel generators to reflect the use of locked-open ERCW valves versus normally closed valves. The changes to the model in the area of high pressure recirculation resulted from crediting the improvement in operator actions resulting from changes to the EOPs such that one train of Containment Spray is stopped, thus increasing the time for RWST depletion. The CCS analysis was revised to include the 2B-B pump in train B. During the revision, systemic errors in the modeling of CCS were corrected. The initiating event frequency for both CCS train A and the total loss of CCS were both revised downward.

- The PSA was revised again prior to the peer review by the Westinghouse Owner's Group (WOG) in April 2001. In addition to documentation updates to meet the requirements of the peer review, this revision incorporated a series of interim model changes that 1) integrated the Level 1 and Level 2 models to allow calculation of LERF; 2) updated the reactor coolant pump (RCP) seal

LOCA analysis to reflect that the RCP seals have been replaced with seals that are qualified for high temperatures; and 3) incorporated changes to ensure that the PSA reflects the as-built, as-operated plant including plant-specific data. The CDF and LERF were calculated as 4.5E-5/reactor-year and 1.65E-6/reactor-year, respectively.

- Revision 4 to the PSA was performed to modify the model in order to supply the information required by the Mitigating Systems Performance Indicator (MSPI) Program. This revision of the model resolved WOG peer review Facts and Observations (F&Os) that were determined to impact MSPI; updated the model to current plant design; updated the initiating event data based on the latest plant-specific and industry data; incorporated the latest maintenance rule data into the database; and incorporated comments on the systems analyses by the WBN system engineers. Also, changes were made to the model to permit calculation of Fussel-Vessely importance values of certain maintenance alignments in support of the MSPI program. The CDF was revised to 1.26E-05 per reactor year. The LERF was revised to 3.31E-07 per reactor year.

c. <u>*NRC Request*</u>

Identify any physical or procedural modification changes to Unit 1 since the release of Revision 4 of the PRA could have a significant impact on the results of the Unit 2 PRA or SAMA analysis. Provide a qualitative assessment of their impact on the PRA and on the results of the SAMA analysis.

<u>TVA Response</u>

No major procedural changes were identified. The following list of modifications is provided:

- The WBN Unit 1 SGs were replaced. This modification should decrease the probability of a SG tube rupture in WBN Unit 1. The WBN Unit 2 SGs are not being replaced prior to Unit operation. In preparation for the development of a dual unit model, an analysis comparing the two types of SGs determined that the the old SGs are bounding. Therefore, there is no impact on the PRA or SAMA results related to this change.

- Modifications to meet the WBN commitments made in response to GSI-191, such as the replacement of the containment sump screens and DCN 52226, which performed banding of Min-K material in containment (Reference SAMA 198), were completed. These modifications would result in a reduction to the CDF and LERF values in the PRA model by decreasing the probability that the containment sump screens would plug, thereby increasing the probability that recirculation would be successful.

- A modification to the ERCW system was performed to allow Headers 1A&2A (or 1B&2B) to be cross-connected by opening cross-connect valves

1-ISV-67-1117 and 2-ISV-67-1119 (or 1-ISV-67-1118 and 2-ISV-67-1120). This modification would result in a decrease to CDF and LERF by allowing continued flow to ERCW headers during strainer maintenance.

- A modification was performed to provide alternate inverters to the vital battery boards. This modification would result in a decrease to the plant CDF and LERF.

d. _NRC Request_

Provide a description of any differences in the designs of Unit 1 and Unit 2 that are expected to exist at the time Unit 2 begins operation. Discuss the estimated impact of these differences on CDF, release frequencies, and the results of the SAMA analysis.

TVA Response

Other than the SGs discussed above, the following are currently known differences between the units:

- Eagle 21 - Unit 2 will use 4-20mA transmitters rather than 10-50mA transmitters used for Unit 1. This change has no impact on the PRA or SAMA analysis
- Currently there are plans to replace various control loops of both units with Foxboro I/A (digital controls). This may not be complete for Unit 1 at the time of Unit 2 licensing. This change has the potential to impact the reliability of components modeled in the PRA or SAMA analyses; however, there is no accurate data currently available on the reliability of these systems to determine impact on the PRA or SAMA analyses.
- Currently there are plans to replace the annunciator systems of both units with an upgraded Ronan system. This may not be complete for Unit 1 at the time of Unit 2 licensing. This change is not expected to impact the PRA or SAMA analyses.
- The Hydrogen Analyzers for Unit 2 will be purchased as non-safety-related. There is no known difference in the reliability of the analyzers; therefore, there is no impact on the PRA or SAMA analyses.
- The Incore probes on Unit 2 will be WINCISE fixed probes as opposed to the Unit 1 Westinghouse Traversing. This change does not impact the PRA or SAMA analyses.
- The Unit 2 main generator voltage regulator will be a new digital regulator. This change is also planned for Unit 1, but may not be completed at the time of Unit 2 licensing. This change is not expected to impact the PRA or SAMA analyses.
- The Unit 2 ice condenser chillers will be replaced with more environmentally friendly units. This change is also planned for Unit 1, but may not be completed at the time of Unit 2 licensing. This change is not expected to impact the PRA or SAMA analyses.

- Unit 2 will eliminate the Post Accident Sampling system. This will not impact the PRA or SAMA analyses.
- The Inadequate Core Cooling Monitoring System-86 will be replaced in Unit 2 with the Common Q platform. This change is not expected to impact the PRA or SAMA analyses.
- There are plans to replace the loose parts monitoring system. This change is also planned for Unit 1, but may not be completed at the time of Unit 2 licensing. This change is not expected to impact the PRA or SAMA analyses.
- There are plans to replace the TEC RCP vibration system with a new design from Bentley Nevada. This change is also planned for Unit 1, but may not be completed at the time of Unit 2 licensing. This change is not expected to impact the PRA or SAMA analyses.
- The turbine and feedwater pump vibration systems for Unit 2 will be an updated Bentley Nevada design. This change is not expected to impact the PRA or SAMA analyses.
- The LEFM for Unit 2 will be an updated Caldon design. This change is not expected to impact the PRA or SAMA analyses.
- ECCS flow balancing for Unit 2 will use throttle valves as opposed to a combination of throttle valves and orifices in Unit 2. The system design is not yet completed, but will have manual valves locked in place and is not expected to increase the CDF for Unit 2 in both the PRA and SAMA analyses.
- Unit 2 will not produce tritium at the time of unit licensing. The Unit 1 timing is the limiting analysis, and the same timing will be used for both units for hot leg recirculation. Therefore, this difference has no impact on the PRA or the SAMA analyses.

e. <u>*NRC Request*</u>

It is stated in Section 4.1 that as a result of discussions with Sequoyah personnel, two PRA model changes were made, and that following a review of the various shared systems with the Sequoyah model, no further model changes related to shared systems were identified as necessary for the Unit 2 SAMA model.

i. Identify the shared systems at WBN.

<u>TVA Response</u>

The shared systems between WBN Units 1 and 2 in the PRA model consist of electric power systems; ERCW; CCS; plant and control air systems; and heating, ventilation and air conditioning (HVAC).

ii. <u>*NRC Request*</u>

Discuss the modeling of shared systems in the SAMA model.

TVA Response

- Modeling for electric power in the SAMA model is as described for the WBN PRA model (i.e., similar to the Sequoyah PRA model). This model is based on the structure developed for the original IPE, which modeled both Unit 1 and Unit 2 buses. These buses supply shared system components, such as ERCW, CCS, and HVAC.
- Modeling of ERCW in the SAMA model is as described for the WBN PRA model. This model is based on the structure developed for the original IPE, which modeled both Unit 1 and Unit 2 pumps and trains. These pumps and trains supply shared system components, such as CCS and HVAC.
- In the PRA model, all five CCS pumps are available to support Unit 1 operation. The SAMA model was revised to reflect 2-unit operation. SAMA modeling of the CCS system has the 1A-A and 1B-B pumps normally aligned to Unit 1 Train A service and the 2A-A and 2B-B pumps normally aligned to supply Unit 2 Train A. The C-S pump is assigned to Train B service for both units, with the 1B-B and 2B-B pumps available to provide B train service when the C-S pump is unavailable due to maintenance.
- Modeling of HVAC in the SAMA model is as described for the WBN PRA model. HVAC systems are modeled as supporting the common electrical switchgear areas (switchgear board rooms, etc.).
- Modeling of plant and auxiliary control air in the SAMA model is as described for the WBN PRA model. The plant and auxiliary control air systems are modeled as supporting components in both units requiring air.

iii. *NRC Request*

Discuss the modeling of dual unit initiating events, including the status of the other unit.

TVA Response

For dual unit initiating events, such as loss of plant compressed air (LOPA) and loss of offsite power (LOSP), it is assumed that both units are initially at power, such that resource requirements for shutdown would be maximized (makeup due to decay heat, need for RHR and SI makeup, etc.).

iv. *NRC Request*

Clarify the relevance to Unit 2 of the first mentioned change, that is, "Changes to the component cooling water system (CCS) to remove credit for the Unit 2 pumps from the Unit 1 model to reflect dual unit operation."

TVA Response

In the PRA model, all five CCS pumps are available to support Unit 1 operation. This initial (Unit 1 only) configuration allowed for the use of both Unit 2 (2A-A and 2B-B) CCS pumps and Unit 1 pumps (1A-A and 2A-A) to be allocated to Unit 1 operation, in addition to the C-S pump. In the SAMA model, this was revised to reflect dual unit operation. When Unit 2 CCS operation begins, the 2A-A and 2B-B CCS pumps will be dedicated to Unit 2 operation, as currently is done with Sequoyah. Modeling of CCS system for dual unit operation has the 1A-A and 1B-B pumps normally aligned to Unit 1 Train A service and the 2A-A and 2B-B pumps normally aligned to supply Unit 2. The C-S pump is assigned to Train B service, with the 1B-B and 2B-B pumps available to provide B train service when the C-S pump is unavailable due to maintenance.

f. *NRC Request*

Provide a description of the Tennessee Valley Authority (TVA) processes for Level 1 and 2 PRA updates, quality control of PRA model changes, and independent review and approval of PRA model update documentation. Include a discussion of the scope, review criteria, and results of any independent reviews of the Level 1 and/or the Level 2 models other than that by the Westinghouse Owners Group (WOG), including any internal or external reviews against the American Society of Mechanical Engineers (ASME) PRA standard (ASME RA-Sc-2007) or Nuclear Regulatory Commission (NRC) Regulatory Guide 1.200.

TVA Response

The TVA process for controlling updates to the PRA is documented in TVA procedure SPP-9.11, "The Probabilistic Risk Assessment Program," and Nuclear Engineering Department Procedure (NEDP) 26, "Probabilistic Risk Assessment." SPP-9.11 covers the management of PRA application, periodic updates and interdepartmental PRA documentation. This procedure provides definitions for PRA model update, PRA model application, and PRA evaluation. This procedure also defines responsibilities of other departments, such as operations and system engineering for review of the PRA. NEDP-26 describes the process used by the PRA staff to perform applications, model updates and PRA model maintenance and review. The terms PRA upgrade and maintenance are defined in the procedures using the definitions provided in the ASME standard. The procedure requires that updates should be completed at least once every other fuel cycle (for the lead unit at multi-unit sites) or sooner if estimated cumulative impact of plant configuration changes exceeds +10% of CDF. Changes in PRA inputs or discovery of new information shall be evaluated to determine whether such information warrants a PRA update. Changes that do not meet the threshold requirement for immediate update are tracked in an open items database. PRA updates shall follow the guidelines established by ASME RA- Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," for a minimum of a Category II assessment. This procedure also defines the requirements for PRA documentation of the model of record (MOR) and PRA applications. The MOR is composed of the

(1) PRA computer model and supporting documentation, (2) MAAP model and supporting documentation, and (3) other supporting computer evaluations (e.g., STADIC, BARP, EPRI HRA Calculator, etc.). The purpose of the PRA MOR is to provide a prescriptive method for quality, configuration, and documentation control. PRA applications and evaluations are referenced to a MOR, and therefore the pedigree of PRA applications and evaluations is traceable and verifiable. After September 2008, modified PRA notebooks were converted to TVA calculations in accordance with TVA procedure NEDP-2. The calculation process requires calculations to be prepared and independently checked and approved. NEDP-26 also specifies the requirements for independent review and periodic self-assessments of the model. A self-assessment was performed prior to the WOG review, but no reviews have been performed on the SAMA model specifically to the ASME standard other than the peer review.

g. <u>NRC Request</u>

Provide the date of the WOG peer review, the PRA model revision reviewed, and the overall results of the peer review as stated in the peer review report.

<u>TVA Response</u>

The WBN peer review was conducted April 21-27, 2001. The peer review team reviewed R3 to the PRA model. The following is quoted from the General Summary section of the report:

"All of the technical elements were graded as sufficient to support applications requiring the capabilities of a grade 2, e.g., risk ranking applications. The Watts Bar Nuclear Plant PSA thus provides an appropriate and sufficiently robust tool to support such activities as Maintenance Rule implementation, supported as necessary by deterministic insights and plant expert panel input.

Most of the elements were further graded as sufficient to support applications requiring the capabilities defined for grade 3, e.g., risk-informed applications supported by deterministic insights, but in some cases this is contingent upon implementation of recommended enhancements. ...The general assessment of the peer reviewers was that the Watts Bar PSA can be effectively used to support applications involving risk significance evaluations supported by deterministic analyses, *once the items noted in the element summaries and Fact & Observation sheets are addressed.*"

A table of specific F&Os were provided in the previous submittal.

h. <u>NRC Request</u>

Table 2 of the SAMA Analysis Report provides the resolution status of the Level A and B WOG peer review Facts and Observations (F&Os). Provide additional justification for the adequacy of the resolution of the following F&Os:

i. SY-08 -This F&O raises a question concerning the applicability of the emergency diesel generator (EDG) repair times to WBN. The resolution states that documentation of the basis is not necessary for the SAMA analysis. Justify the EDG repair times used in the SAMA model or that they would not impact the results of the SAMA analysis.

TVA Response

The intent of the response was that TVA considered this F&O a documentation issue and that correcting the documentation was not necessary prior to the SAMA analysis. The WBN model is a RISKMAN model and the Electric Power Recovery model reflects recovery factors developed and computed for time averaged conditions using the STADIC code and methodology used by PLG (now ABS consulting). For risk monitoring purposes, the model recognizes that specific diesel generators could be undergoing maintenance at the time of the LOSP and that these diesel generators would not be considered recoverable. Recovery times influencing the diesel generators specifically are time following operator response (which includes detection time, notification time, and transit time); frequency of hardware recovery; the time interval between the beginning of SBO; and the point of no return of the electric power system prior to core damage.

ii. *NRC Request*

DE-02 -This F&O discusses the adequacy of the internal flooding analysis. The F&O notes that portions of the analysis are based on engineering judgment or are not supported, and recommends a number of actions including reviewing the analysis against RI-ISI and HELB/MELB analyses. (The 2001 WBN RI-ISI submittal indicates a CDF of 4.6E-05 per year for pipe failures in Class 1 and 2 systems.) The resolution does not address these recommendations. Justify that the issues raised in the F&O would not impact the results of the SAMA analysis.

TVA Response

The flooding analysis is judged to conservatively bound the contribution due to these contributors, such that a rigorous re-evaluation of the flooding for the SAMA effort was judged to be unnecessary. In the analysis of auxiliary building flooding discussed in the F&O, pipe breaks on upper elevations would propagate down stair wells and through grated flooring to the lowest elevation in auxiliary building and the passive sump. Operator actions to isolate piping were assumed to have the same isolation timing regardless of size. A more rigorous analysis as suggested by the F&O resolution judged to result in less conservative consequences and therefore, the existing analysis was judged to be bounded by the results of the SAMA analysis.

iii. <u>NRC Request</u>

HR-01 and HR-15 -These F&Os question the treatment of human action dependencies. The resolution indicates that a sensitivity study was performed wherein operator actions were set to guaranteed failure and the resulting top 50 sequences were reviewed to identify necessary model changes. Based on this review, a number of top events were set to guaranteed failure. Justify that review and treatment of the top 50 sequences is sufficient to uncover all significant human actions that could be impacted by potential SAMAs. Explain how setting certain top events to guaranteed failure properly treats dependent human actions and does not hide important actions that might be the source of a potential SAMA.

<u>TVA Response</u>

Setting the operator action related top events to guaranteed failure artificially increases all scenarios in which the subject top event is questioned. This reveals scenarios in which multiple operator actions were independently questioned (i.e., with a more reliable, non-dependent failure rate that which should have been used). During this review, seven groups of operator actions were identified in which dependency between operator actions should have been more conservatively addressed (i.e., represented by a higher failure rate than currently shown in the base PRA model). Setting subsequent operator actions to guaranteed failure then increases the total scenario frequency, reflecting increased dependency between previously independent operator actions. This was judged adequate to properly reflect the impact of operator actions for the SAMA model.

The scenarios reviewed through this exercise had frequencies of 1.3E-2 to 9.8E-5, such that a broad spectrum of potential dependent actions was revealed. This steep form of risk profile allows the capture of a large number of dependent action families in a relatively small (i.e., 50) number of top scenarios.

iv. <u>NRC Request</u>

HR-11 -This F&O raises an issue regarding the consistency of time estimates used in establishing the human error probabilities (HEPs). The resolution: indicates that the human reliability analysis (HRA) was updated to use the Electric Power Research Institute HRA calculator, specifically addresses the two operator actions mentioned in the F&O, and states that the F&O can be considered closed if the revised time windows for these two operator actions are used. Provide additional information regarding the statement that shortening the time window for bleed and feed (from 30 to 10 minutes with only a safety injection pump available) does not change the resulting HEPs. Justify why the time windows for additional operator actions should not be reconsidered consistent with this F&O recommendation, and why this would not impact the results of the SAMA analysis, that is, the identification of additional SAMAs involving procedure improvements.

TVA Response

The Resolution Status states that "the resulting HEP did not change substantially" (3.00E-3 for OB1 and 3.40E-3 for OB2). That is, for the two cases discussed, the operator has adequate time to diagnose and perform the action and the amount of time to irreversible consequences is (and stays) long relative to the time to diagnose and perform each of these actions.

As noted in the response, based on engineering judgement the two actions identified are the two most time-sensitive in the PRA, such that these represent a bounding case argument for not requiring comprehensive update of all operator actions for the SAMA model.

v. *NRC Request*

L2-03 and L2-05 -These F&Os note that the Level 2 analysis does not include operator actions and that the reliance on NUREG-1150 analyses could impact the Level 2 results. The resolution for both F&Os indicates that the current model results in a conservative offsite consequence and maximum possible benefit. However, inclusion of operator actions in the Level 2 model or updating the model to reflect recent research information could conceivably lead to the identification of additional candidate SAMAs, particularly enhancements to procedures and guidance to improve operator response following core damage. Identify key operator actions that are amenable to treatment within the Level 2 analysis, and discuss their potential impact on the results of the Level 2 and SAMA analysis. Provide an assessment of potential SAMAs related to these actions.

TVA Response

Identified actions for L2-03 are (1) proceduralize operator action to depressurize the reactor coolant system (RCS) after core damage has occurred per FR-C.1, (2) manually open AFW discharge valves after loss of all instrument air, and (3) SAMG actions. Each of these is currently identified in the plant as a response action in the given situation, such that "new" SAMAs that could be identified are judged to already be included in the suite of EOP and SAMG actions in place at WBN.

Level 2 SAMA items are well represented by SAMAs 92-101 and 110. SAMA 103 specifically addresses simulator training for severe accident response.

vi. *NRC Request*

TH-06 and TH-10 -These F&Os question the bases for the success criteria used for small LOCA and bleed and feed. The resolution indicates that success criteria analyses were performed for the WBN Unit 2 PRA. Describe the scope of the analyses, the computer code(s) used, and the results of these analyses.

TVA Response

The success criteria for maintaining core cooling for small LOCA and transient events with the subsequent loss of all AFW was determined using thermal hydraulic analysis results from the MAAP4.0.5 code using WBN plant design features. The success criteria consists of three separate, but inter-related parameters: the minimum number and type of high pressure Emergency Core Cooling (ECC) pumps available, number of pressurizer PORVs available for each high pressure ECC pump configuration, and time window available for successful implementation of the bleed and feed core cooling mode for each configuration.

The analyses identified that the high head ECC pump configuration is of high importance in determining the overall success criteria. For the WBN design, the charging pump can deliver some flow when the RCS pressure is at the pressurizer PORV setpoint. Thus, the number of pressurizer PORVs is minimized and the time window for implementing bleed and feed cooling is maximized compared to the configuration with all charging pumps failed. With only the high head SI pumps available, the RCS must be depressurized below the pump shutoff head in order for injection flow to be effective for core cooling, which translates to a requirement for greater relief capability and smaller time window for successful operator action to implement bleed and feed cooling.

Three different initiating event types were analyzed, based on unique core cooling challenges presented by each. For the small LOCA events, some decay heat can be removed by break flow so that less relief capacity through the pressurizer PORVs is required compared to non-LOCA events. The loss of main feedwater initiating event is a unique non-LOCA event because the reactor trip occurs on low SG level as opposed to reactor trip on other signals for the other non-LOCA events. Thus, for the loss of all feedwater event (loss of main feedwater followed by unavailability of AFW), a smaller SG inventory is available for initial decay heat removal, which shortens the time at which bleed and feed cooling must be initiated.

The minimum success criteria for each unique set of initiating events and configuration of ECC pumps is provided in the table below.

1. WBN Feed and Bleed Success Criteria

Initiating Event	Centrifugal Charging Pump	Safety Injection Pump	Power Operated Relief Valve	Latest Time to Initiate Bleed and Feed(1)
Large and Medium LOCA	**Bleed and Feed is not required.**			
Small LOCA (Break size < 2 inches and > 3/8 inch)	**1**	**0**	**1**	**Within 1 hour**
	0	**1**	**2**	**Within 25 minutes**
Transients, except loss of all feedwater	**1**	**0**	**1**	**Within 45 minutes**
	0	**1**	**2**	**Within 15 minutes**
Loss of All Feedwater (loss of main feedwater initiator)	**1**	**0**	**1**	**Within 25 minutes**
	0	**1**	**2**	**Within 10 minutes**
(1) All times are referenced to the time at which the SG level drops to less than 26% Wide Range SG level span, which is the criterion for initiating bleed and feed in the WBN FR-H.1, "Response to Loss of Heat Sink" emergency operating instruction.				

The MAAP4.0.5 analyses were compared to the results of the most recent analyses in WCAP-16902-P. These analyses were based on the licensing basis NOTRUMP computer code but incorporated best estimate input values. The 4-loop reference plant with 51 Series SGs is the closest to the WBN plant design for this comparison. Considering differences in trip setpoints and ECC capabilities between WBN and the WCAP-16902 reference plant, the MAAP4.0.5 results are judged to be reasonable.

It should be noted that two pump flows for WBN do not significantly improve the success criteria for the case with only one safety injection pump available.

The new success criterion for bleed and feed is more relaxed than the WBN Revision 4 PRA model in terms of equipment requirements but more restrictive in terms of timing for the operator action. The Rev. 4 model required 2 of 2 pressurizer PORVs and 1 of 4 high head pumps for success. The new success criteria require only one PORV if at least one charging pump is available. The Rev. 4 model time window for successful operator actions HAOB1 and HAOB2 is 50 minutes after the 25% wide range SG level is diagnosed whereas the new success criteria is 25 minutes if at least one charging pump is available and 10 minutes if only an SI pump is available. However, substituting the new times into the HRA calculator tool used for the Rev. 4 HRA model shows that the human error probabilities do not change, because multiple recovery actions are assumed to be available in the current HRA model.

vii. NRC Request

TVA-001, TVA-002 and TVA-11 -These F&Os appearing in Table 2 are apparently from an internal self-assessment rather than the WOG peer review. Discuss the source of these items and identify any other items from this source that might be relevant to judging the quality of the SAMA PRA.

TVA Response

TVA maintains an open item database to ensure that all items impacting the PRA are tracked and appropriately resolved. This database includes WOG peer review F&Os as well as issues found during TVA's internal reviews. As noted in the description of TVA-001 and TVA-002, these were identified during a self-assessment effectiveness review effort, as described in Report WBN-ENG-01-005. There were no other items identified through this effort that were judged to impact the quality of the IPE/PRA model.

TVA-11 was identified during a review of risk significant components as part of the preparation for a Comprehensive Design Basis Inspection (CDBI). Again, no other items identified through this effort were judged to impact the quality of the IPE/PRA model.

i.[sic] NRC Request

In the discussion of the individual F&Os on page 5, the last bullet mentions a potential change to the SAMA model associated with ventilation system recovery. However, a model change involving ventilation system recovery is not identified in Table 2. Confirm whether the ventilation system recovery change was made in the SAMA model.

TVA Response

TVA provided the findings in Table 2. As part of the SAMA effort, the observations from the WOG report were also evaluated. Based on the above evaluation, ventilation recovery was identified as a potential impact to the SAMA model as discussed on page 5. An additional outcome of the evaluation is that the impact of the ventilation dependency remains in the model. See SAMA 278.

2. NRC Request

Provide the following information relative to the Level 2 PRA analysis:

a. The Level 2 PRA is stated to utilize containment event trees (CETs) developed for the individual plant examination (IPE). The IPE utilized 26 plant damage states (PDSs) that were later collapsed into 10 key PDSs based on PDS frequency and the IPE guidance document reporting requirements. Clarify if this *same process was applied for the SAMA analysis. If not, discuss how the process was performed for the SAMA Level 2 model, the general results of the analysis, and the impact on release category frequencies and source terms.*

TVA Response

The WBN Unit 1 IPE CET model was imported into the SAMA model through the following process:

1. Restore archived Riskman Advanced Revelation based models that performed IPE and release category binning to Riskman for DOS.
2. Migrate CET from this model to Riskman for Windows.
3. Reproduce the CET structure in SAMA model.
4. Migrate split fraction rules to new model. This includes adjustment of split fraction designations for conflicting top event designations noted below.
5. Migrate interim variable (Macro) assignments to new model. This includes designation of new macros for all previous CET initiating events.
6. Migrate release category binning logic to new model. Again, this includes adjustment of top event designations to prevent conflict with Level 1 model top event designations.
7. Translate new model from PDS initiator basis to establish Level 1 model conditions for each PDS. In general, this can be seen in the INIT=ENS form to using new macro PDSENS to establish the model conditions.
8. Incorporate CET module into Level 1 initiating event logic.
9. Debug model structure for remaining Level 1 model structures (MLOCA, LLOCA, VS LOCA).

This process resulted in an integrated Level 1/Level 2 plant model that progressed through the original PDS assignments, as described in Step 7 above, and directly generated the release categories associated with the given plant model scenario.

b. *NRC Request*

It is stated in Section 4.6.4 that the CET nodes and split fractions of NUREG-1150 were reviewed to assure that the consequences in terms of release frequencies would be larger than would be expected with an updated Level 2 model. Provide additional discussion of how this was done, what adjustments were made, if any, and the source of any updated values.

TVA Response

Section 4.6.4 states that the WBN Level 2 model is represented by a large CET that is based on the NUREG-1150 assessment for SQN. This statement was provided to show the history of the WBN Level 2 model from the development of the WBN IPE. The NUREG-1150 model was used as the basis for the WBN model. The event tree nodes and split fractions in the WBN Level 2 model were reviewed to ensure that the release frequencies were adequate due to an issue identified in the WOG peer review. Specifically, Revision 3 of the WBN plant risk model included a Level 2 top event, CV, which was used to "adjust" the core damage frequency for certain Level 1 end states to address in-vessel recovery of core damage for scenarios that should not have been assigned to core damage. In particular, the Level 1 model previously assumed that two PORVs were required for successful bleed and feed cooling when charging pumps were available. See response to RAI 1h.vi above for current analysis results for bleed and feed success criteria.

As shown in the table below, this in-vessel recovery issue was resolved in the Revision 4 update. This can be seen from the LERF contribution for endstates LNIYA and LNIYC, which went from all CDF arrested in vessel to over 93% resulting in a large early release for endstate LNIYA and over 16% for end state LNIYC.

Core Damage Arrest Top Event CV

Level 1 Endstate (PDS)	WBN R3 Top Event CV Value (CDF Recovered by In-Vessel Arrest)	WBN R4 LERF Top Event Value
BCI (MLOCA with no containment sump recirc)	**0.77 (23% recovered)**	**0.008 (LER7)**
FCI (SLOCA with no containment sump recirc)	**0.68 (32% recovered)**	**0.008 (LER7)**
LCI	**0.23 (77% recovered)**	**0.155 (LER5)**
LNIYA	**0.00 (All recovered – no CDF)**	**0.933 (LER1)**
LNIYC	**0.00 (All recovered – no CDF)**	**0.166 (LER4)**

c. *NRC Request*

Section 4.2 states that the release categories were retained from the IPE Level 2 model and the binning of release categories (RCs) into four categories was also retained from the IPE. RC II is defined in the IPE update as small early containment failures and small bypasses, including SG tube rupture (SGTR) events. However, the iodine release fraction for RC II in the SAMA analysis (Table 7) is 0.21, which is not "small." Table 4.2-1 of the IPE update indicates that the frequency of RC II is 10 percent of the update CDF. However, the frequency of RC II in the SAMA submittal (Table 3) is less than 1 percent of the SAMA CDF. Discuss the development of RC II and why its magnitude and frequency is significantly different from the IPE.

TVA Response

The size (large/small) is relative to the size of the containment breach, not the relative fraction of isotopic release (although the two could be related). In the case of RC II, the primary contributor has historically been SGTR, which dropped from a 5% contribution to CDF in the IPE update to 0.42% contributor (see response to RAI 1a, above) in the SAMA model. This is due to refinement in the modeling of plant and operator response to SGTR.

d. *NRC Request*

Section 4.6.4 refers to the "WBN2 Level 2 Model." Describe this model and how it differs from the IPE (Unit 1) Level 2 model. Clarify whether the "WBN2 Level 2 model" is the Level 2 portion of the WBN4SAMA model.

TVA Response

The WBN2 Level 2 model is the Level 2 portion of the WBN4SAMA model. The latest revision to the WBN Unit 1 IPE Level 2 model consists of an integrated event tree model that takes the equivalent plant damage states (as represented by Level 1 top event success or failure) as input and assigns the probabilistic value of LERF/NOLERF results. Conversely, the model developed for the SAMA model integrates the actual CET from the initial IPE model into the Level 1 model, such that the end states are the associated release categories for the various model scenarios rather than PDS (LERF/NOLERF).

e. *NRC Request*

Tables 4 through 8 reference[s] a 2007 analysis by Science Applications International Corporation (SAIC) entitled "Watts Bar Nuclear Plant Severe Reactor Accident Analysis." Describe the relationship between the SAIC analysis, the "WBN2 Level 2 model," and the Level 2 portion of the WBN4 SAMA model, and how the release characteristics used in the SAMA analysis were developed. Provide a copy of the SAIC report.

TVA Response

The SAIC report was generated in 2007 to support the Supplemental Environmental Impact Statement for WBN to include Unit 2. This was based on the CET model developed to support the initial IPE for WBN and uses Level 2 information from that report (see discussion of nuclide distribution from the IPE under RAI 2f, below), as was the CET portion of the WBN4SAMA model. As noted in the response to RAI 2d above, the Level 2 model currently being used generates LERF and NOLERF endstates, rather than release categories, as done for the SAMA model.

An electronic copy of the referenced SAIC report is attached.

f. *NRC Request*

The radionuclide release characteristics (I and Cs release fractions, release time and release duration) provided in Tables 6 and 7 of the SAMA submittal are, for some release categories, significantly different from those given for Sequoyah in Table 3.10 of NUREG/CR-6295. Discuss the reasons for these differences and justify the values used in the SAMA analysis.

TVA Response

The values used in the SAMA analysis (and the SAIC report cited in RAI 2e) were taken from the plant-specific Level 2 analysis performed for the original WBN IPE. These values were taken from Table 4.9-5 for RC I, Table 4.9-9 for RC II, and Table 4.9-11 for RC III. The source terms for each of these evaluations were developed by different analytical teams using different analytical codes (SEQSOR code for NUREG/CR-6295 and MAAP 3B for the IPE) which resulted in different time and release characteristics. The IPE used a parameter file for the analysis that was specific to the WBN plant. Also, there is some variation noted in the time of release and release duration used in the analysis.

3. *NRC Request*

Provide the following information regarding the treatment of external events in the SAMA analysis:

a. *The NRC staff safety evaluation report (SER) on the Unit 1 Individual Plant Examination of External Events (IPEEE) noted two issues under the Multiple System Response Program (MSRP/GSI [Generic Safety Issue]-172) for which the staff could not find complete information in the licensee submittal. The first issue was non safety-related control system/safety-related system dependencies; the second was the effect of flooding and/or moisture intrusion for nonsafety-related equipment. Describe the resolution of these two issues for Unit 1 and the implications for the Unit 2 SAMA analysIs.*

TVA Response

A review was performed to determine the resolution of the two issues noted in the Unit 1 IPEEE report. The SER (TAC NO. M83693) noted that "...these individual MSRP issues for WBN will be addressed by the NRC staff separately from the IPEEE program." Based on the review performed, no documentation could be found to show that these two items were resolved by the Staff following approval of the IPEEE for Unit 1. The following addresses these two issues.

The description of GSI-172 in NUREG-0933 discusses that multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems as a result of potential unrecognized dependencies between control and protection systems.

As stated in the Unit 1 IPEEE (Attachment 4 Section 7.2.5) response to the resolution of Sandia Fire Risk Scoping Study on control systems interactions for WBN Unit 1, "The remote shutdown system at Watts Bar consists of the Auxiliary Control Room and shutdown boards that are located in the Auxiliary Building. The remote shutdown system circuits are physically independent of, or can be electrically isolated from, the Main Control Room. Therefore, safe shutdown can be accomplished from outside the Control Building in the event of a severe fire in the Control Building that would cause Main Control Room abandonment. This capability is described in Part IV of the Fire Protection Report. The implementation of this capability is directed by Appendix C.69 of Abnormal Operating Instruction 30.2, 'Fire Safe Shutdown.'"

The IPEEE submittal for WBN Unit 2 was submitted to the NRC on April 30, 2010, and addressed this issue of control systems interaction similarly. Additional information is also provided on control systems in IPEEE Section 7.3.7, which states "For this analysis, control systems were assumed to fail in such a way as to fail the function of the affected system. It should be noted that this analysis conservatively assumes that 'hot short' failures occur whenever necessary to fail the system function."

Flooding and/or water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures, actuations of fire suppression system, or backflow through part of the plant drainage system. The IPE submittal guidance (Generic Letter 88-20 and NUREG-1335) includes consideration of moisture intrusion and internal flooding. The IPE/IPEEE process should detect plant-specific vulnerabilities identified in the ACRS concern.

b. *NRC Request*

The Unit 1 IPEEE provides the fire CDF for the dominant fire areas. All fire areas have a fire CDF less than 1.0E-06 per year. Clarify whether any fire areas at Unit 2 *are expected to be substantially different than those reported for Unit 1 and describe the reasons for any differences.*

TVA Response

The development of a design basis IPEEE for Unit 2 has been completed and was submitted to the NRC in a letter dated April 30, 2010. The CDFs calculated for Unit 2 Fire Areas are below 1E-06 and have not been shown to be substantially different than those provided for Unit 1.

c. *NRC Request*

TVA used an external events multiplier of 2.0 in the SAMA analysis based on a review of other SAMA analyses. This multiplier could be considerably higher based on the plant-specific seismic and fire CDF. The NRC staff notes that the WBN seismic CDF would be about 5E-05 per year using the simplified hybrid method for estimating seismic CDF (Kennedy, R. P., 1999 "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop of Seismic Risk, Tokyo, Japan 10-12 August 1999), the latest U.S. Geological Survey seismic hazard curve, and a WBN plant fragility (high confidence low probability of failure) of 0.36g. Use of this seismic CDF in conjunction with the IPEEE fire CDF of 7E-06 per year would result in an external events multiplier of 4.7. Provide an assessment of the impact on the Phase I and II SAMA results (baseline and baseline with uncertainty) based on an external events multiplier of 4.7, or justification for use of a lower multiplier. This assessment can be limited to internal event SAMAs that could have significant benefits in external events.

TVA Response

TVA believes the calculation performed to determine the seismic CDF of 5E-05 is incomplete. Seismic CDF in an IPE would be calculated using seismic hazard curves and uniform hazard spectra at various annual exceedance probabilities. The IPEEE HCLPF values for WBN are based on a deterministic response spectrum developed by multiplying the original design basis Modified Newmark Spectrum anchored at a peak ground acceleration (pga) of 0.18g by a factor of 1.67 to provide the Review Level Earthquake (RLE) anchored to a pga of 0.30g. The IPEEE spectrum is, therefore, not directly comparable to a uniform hazard shape. To develop seismic CDF from IPEEE HCLPF values, it is necessary to account for this difference in spectral shape. This was recognized by the NRC when the closure evaluation was performed for Generic Issue (GI) 194 for WBN. Figure 1 from GI-194 closure evaluation provides the additional computation performed to account for the difference in spectral shape. In the GI-194 closure evaluation, the NRC used the same method developed by R. P. Kennedy as cited in the question (see Reference 9 of the NRC document) to determine seismic CDF for WBN using the seismic hazard data from a Trial Implementation Plan (TIP) in 1998 (see the third paragraph of the section entitled "Risk Implication" in the NRC document). An initial value of seismic CDF of 4E-05/ry was computed in the same manner described in the question. The seismic CDF of 4E-05/yr using seismic hazard information from TIP is analogous to the value of 5E-05/yr in SAMA RAI Question 3.c using seismic hazard information from USGS. Further computation to account for differences in spectral shape between the IPEEE spectrum and the TIP uniform hazard spectrum resulted in a seismic CDF of 1.8E-05/ry. TVA believes that computation of an analogous value of averaged CDF, or CDF at a single reference frequency such as 5 Hz, to compensate for the difference in spectral shape between the IPEEE spectrum and a uniform hazard spectrum yields a more appropriate measure of seismic CDF for use in SAMA evaluations. The parameters supporting the complete computation are shown at the end of Attachment 2 of the NRC document. TVA believes that a similar calculation of seismic CDF using the appropriate USGS uniform hazard spectrum would yield a much lower value than 5E-05 cited in the question, and the external events multiplier would be reduced accordingly.

4. NRC Request

Provide the following information relative to the Level 3 PRA analysis:

a. *Identify the computer programs and databases used to determine the population distribution described in Section 4.6.2. Provide additional information on how the population growth rates and the transient population data were developed, including the source of the transient population estimate and how the growth rate estimates were applied.*

TVA Response

No computer programs were used to assemble the population distribution data. Population estimates for 2000 by distance and direction from the WBN site were developed using population counts from the 2000 Census of Population (http://factfinder.census.gov/home/saff/main.html?_lang=en). A map was prepared displaying county and census tract boundaries for all counties partly or totally within the 50 mile boundary. Concentric circles and radii were overlaid on this map to display the 160 segments encompassing the 16 directional segments, divided by distance from the site (1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 miles). County population data for 2000 were allocated to the appropriate sectors, using census tracts to the extent feasible. Block groups were used when census tracts crossed the circles or the radii. In some cases, individual blocks were used as secondary information in determining the allocation. For segments near the plant site, especially within 5 miles, aerial photos and staff knowledge of the area were also used. Projected county growth rates by Woods & Poole Economics, Inc., a firm which specializes in detailed county population and economic projections, were used to project segment populations out to 2030. The county projections were then extended to 2060, using a linear trend line; these projected trends were then used to project the individual segments in each county to 2040, 2050, and 2060.

Transient populations are peak recreation visitation estimates at the various user sites around the Tennessee River system, based on recreation facilities use data.

(These locations account for almost all of the transient population within the 10 mile area.) The transitory (recreation) numbers are based on TVA recreation facility information (internal files), and assume a maximum usage of facilities in the area. The visitation data were extended to 2060 by using the population projections for an eleven-county region around the site. Most of the visitors at these sites are from this region, which includes the cities of Chattanooga and Knoxville. *Source: J.Fouse, S. Melton and Watershed Team Staff; Chickamauga and Watts Bar Reservoirs Recreation Inventory, (spreadsheet data layers), 2006.*

b. _NRC Request_

Identify the version of the SECPOP2000 code used to develop the economic data for the area surrounding the WBN site. If the calculations are not based on Version 3.13.1 of the SECPOP2000 code, confirm that all three recently discovered problems in earlier versions of SECPOP2000 have been accounted for in preparing the WinMACCS2code input for WBN (i.e., a formatting problem in input block text files), an error in formatting the economic database used by SECPOP2000, and gaps in the economic database file. Also provide a discussion of the escalation factors applied to account for changes from the date of the data source to the present.

TVA Response

SECPOP2000 Version 3.12 was used for the MACCS2 Site Data File sent previously for WBN. Neither TVA nor its subcontractors which performed the calculations using SECPOP2000 Version 3.12 were aware of the errors discussed in the NRC question. Consequently, TVA is evaluating the impact of the SECPOP2000 errors and will provide an update (either revised information using SECPOP2000 Version 3.13.1 or an assessment of the impact on the results) by August 31, 2010.

For the calculations which were prepared previously, in order to account for inflation, economic values were escalated (1990 to 2007) using the following method. The year 1990 was assumed to be the benchmark of all dollar input values, the CHRONC input file was compared with sample data that references NUREG/CR-4691, and the values were identical.

All of these input values are in units of dollars per human or land unit. The original data was assumed to be based on 1990 values. To adequately project the dollar values to the year 2007, an inflation factor of 1.555 was used to multiply each of these values.

U.S. Department of Labor annual consumer price index (CPI) values per year from 1990 to 2007 for the Midwest region was used to calculate the inflation factor. The 2008 annual value was not available due to this analysis being performed before the end of the calendar year 2008. The CPI is a measure of the average price of consumer goods and services purchased by households. The percent change in the CPI is commonly used for inflation. The CPI can be used to index (i.e., adjust for the effects of inflation) wages, salaries, pensions, or regulated or contracted prices. To find the correct multiplier to represent the cost increase from 1990 to 2007: Annual CPI 2007 / Annual CPI 1990 = 198.12 / 127.4 = 1.555. The multiplier 1.555 was then applied to all costs that needed to be updated from 1990 to 2007. For example, the value of the input variable CHEVACST001 was updated as follows, CHEVACST001 in 1990 = 27.00 dollars/person-day, 27.00×1.5551 = 41.9877 , therefore, CHEVACST001 in 2007 = 42.00 dollars/person-day. All other identified costs were updated in this manner. This same process will be used for the new calculations using SECPOP2000 V3.13.1.

c. _NRC Request_

Crop production parameters (e.g., fraction of farmland devoted to grains, vegetables, etc.) are stated to have come from the SECPOP2000 for which crop production parameters are based on 1997 National Census of Agriculture. Discuss why information on regional crops was not based on the more up-to-date 2002 Census of Agriculture, and any important differences between these two sources.

TVA Response

The revised SECPOP2000 calculations for the above NRC Question 4.b will be based on updated 2002 Census of Agriculture data.

d. _NRC Request_

Provide the basis for the radionuclide inventory provided in Table 5, including the time in the fuel cycle on which the inventory is based, the computer code utilized to estimate the inventory, and the assumed core power level and burnup. Confirm that this core inventory reflects the expected fuel management/burnup during the license period.

TVA Response

This radionuclide inventory is based on calculations with the SAS2H/ORIGEN-S analysis sequence of the SCALE-4.2 code system. Cross sections for 13 actinides, 181 fission products, and 5 light elements from the SCALE-4.2 burnup library of neutron cross sections (27BURNUPLIB) were collapsed with XSDRNPM-S-derived spectra. The assumed core power level was 3,565 MWt with an initial uranium-235 enrichment of 5 weight per cent and a burnup of 1000 EFPD. The expected WBN2 fuel management (i.e., core power, uranium-235 enrichment and burnup of nuclear fuel in the core) is bounded by that assumed in the calculation of core radionuclide inventory. Therefore, the inventory used in the MACCS2 analysis reflects the expected fuel management and burnup during the license period.

e. *NRC Request*

Identify the source of the meteorological data used in the MACCS2 code (e.g., the onsite meteorology tower). Describe the process used to fill in the gaps for any data missing from the site instrumentation.

TVA Response

The meteorological data used in the MACCS2 code was provided by the onsite WBN Plant meteorological tower that is located 0.85 kilometers south-southwest of the plant. This tower measures wind speed, wind direction, temperature, dew point, solar radiation, and rainfall from instrumentation located at several elevations. All data is collected in accordance with NRC and ANSI/ANS guidance. Gaps in data missing from the site instrumentation at the meteorological tower were filled in using linear interpolation from the recorded data.

f. *NRC Request*

Discuss how precipitation is modeled in the MACCS2 analysis, including the source of the precipitation data, and any assumptions for applying the data (such as forced rain events in the outermost radial ring).

TVA Response

The source of precipitation data is the WBN meteorological tower, which reports hourly rainfall totals. The hourly rainfall data from the meteorological tower is part of the data file used by MACCS2. Five years of data were obtained from this tower, and the year with the largest calculated dose was used in all calculations presented in the ER. The MACCS2 model includes boundary weather data input that applies a large rainfall rate in the last spatial interval of the model which is 40 to 50 miles. This large rainfall boundary condition results in a conservative deposition of radionuclide particles.

5. *NRC Request*

Provide the following information with regard to the selection and screening of Phase I SAMA candidates:

a. *Section 6.2 of the SAMA submittal discusses 12 potential enhancements and insights/recommendations from the original Unit 1 IPE, each of which was included as a candidate SAMA. Section 6.4 of the Unit 1 IPE submittal update includes 13 different additional insights and recommendations. Describe the status of these additional items. For those items that have not been implemented, justify not considering the items as Phase I SAMAs.*

TVA Response

The following Insights and Recommendations were taken from Section 6.4 of IPE Update (the status for each is in italics):

1. Simplify MG Set breaker operation – Add capability to trip RPS MG sets from control room. *This item was considered as a Phase 1 SAMA. Reference SAMA 136*

2. Utilize containment spray pumps for ECCS recirculation. *This item was considered as a Phase 1 SAMA. Reference SAMAs 31, 32 and 33*

3. Enhance procedures with reference to refill of RWST. Refill RWST if ECCS suction is lost. *This item is considered already implemented.* Reference SAMA 33.

4. Alternate power to motor driven MFW pump – consider power from Unit 2 shutdown board and bypass of feedwater isolation under emergency conditions. *This item was considered as a Phase 1 SAMA. Reference SAMA 78*

5. Change TD AFW pump discharge valves to fail open on loss of control air. *AFW flow control reliability is addressed by Phase 1 SAMAs 70 and 223.*

6. Automatic reactor/RCP trip on high RCP motor bearing temperature. *Improved procedural guidance is discussed in SAMA 49. Additional reactor and RCP trip introduces concern for inadvertent plant trips and has significant design and potential safety basis impact.*

7. Alternate cooling (firewater) and power to positive displacement charging pump to reduce impact of loss of RCP seal cooling. *This item was considered as a Phase 1 SAMA. Reference SAMA 215*

8. HVAC procedures – integrated procedure or set of procedures to address loss of HVAC. *This item was considered as a Phase 1 SAMA. Reference SAMAs 160 and 161*

9. Firewater cooling to CCPs – reduces impact of loss of all ERCW. *This item was considered as a Phase 1 SAMA. Reference SAMA 64*

10. Install new (high temperature) RCP seals. *This item was not considered as a SAMA because high temperature RCP seals are included in the design at WBN Unit 2 and therefore considered already implemented.*

11. Make 5th diesel generator operational. *This item was considered as a Phase 1 SAMA. Reference SAMA 9*

12. Cross-tie capability from Unit 2 to Unit 1 6.9 kV shutdown buses – capability was not installed as of IPE update. *This item was considered as a Phase 1 SAMA and is considered already implemented. Reference SAMA 11*

13. Dedicated Connection Capability from Watts Bar Hydro Plant to 6.9KV Shutdown Boards. *Because the existing 161-kV lines originate from the Watts Bar Hydro Plant switchyard, consideration was given in a Phase 1 SAMA to an underground line. Reference SAMA 13*

14. 5th Battery Bus. *This is already implemented at WBN.*

b. *NRC Request*

The 1994 Unit 1 Severe Accident Mitigation Design Alternative (SAMDA) analysis identified 31 potential enhancements that were subjected to further analyses (26 identified by TVA plus 5 identified by NRC staff). While some are included in the current Phase I SAMA list, many are not. Describe the status of each of these enhancements. For those items that have not been implemented or have not been included on the Phase I SAMA list, justify not considering the items as Phase I SAMAs.

TVA Response

The following enhancements were taken from Table 5-1 of the SAMDA report:

I – Improve availability of ECCS recirculation
1. Procedure to stop one train of sprays - This change has been implemented.
2. Install containment spray throttle valves – Reference SAMA 106
3. Redesign to delay containment spray actuation – Reference SAMA 105
4. Install automatic high pressure recirculation – Reference SAMA 32

II – Improve availability of AC power
1. Procedure change to facilitate crosstie of 500kV and 161kV AC power (applicable to Unit 1 only during Unit 2 operation the 500kv line currently used will be dedicated to the Unit 2 generator output)
2. Accelerate availability of fifth emergency diesel generator – Reference SAMA 9
3. Procedure change and fifth diesel – Reference SAMAs 9, 12 and 13

III – Improve ability to cope with loss of AC power and station blackout
1. Procedure change to utilize existing spare 6900V/480V transformers - This was considered applicable to Unit 1 operation only - with 2 unit operation spare transformers will be minimized.
2. Install improved RCP seals (high temperature RCP seals currently installed at Unit 1 and part of the design for Unit 2)

3. Install independent RCP seal cooling system – Reference SAMAs 55 and 56
4. Install accumulators for turbine driven AFW pump flow control valves – Reference SAMA 70
5. Provide DC load shed analysis and procedure. This SAMA is already implemented.
6. Provide portable battery charger (performed as part of B5B effort)
7. Install AC independent coolant injection system – Reference SAMA 28

IV – Improve ability to cope with loss of RCP seal cooling

1. Install improved RCP seals – duplicate of III.2 -Refer to answer provided above.
2. Install independent RCP seal cooling (w/o new EDG) – duplicate of III.3 - Refer to answer provided above.
3. Modify charging pump cooling from CCS to ERCW (implemented for charging pump A) – Reference SAMA 262 for charging pump B

V – Improve containment performance

1. Install deliberate ignition system – hydrogen igniters installed
2. Install reactor cavity flooding system – Reference SAMA 90
3. Install filtered containment venting system – Reference SAMA 94
4. Install core retention device – Reference SAMA 97
5. Install containment inerting system – Reference SAMA 96
6. Install additional containment bypass instrumentation – Reference SAMA 111 and 112
7. Install reactor depressurization system – Reference SAMA 41
8. Install independent containment spray system – Reference SAMA 91
9. Install AC independent air return fan power supplies – Reference SAMA 167

VI – Miscellaneous

1. Install MG set trip breakers in control room – Reference see SAMA 136
2. Improve procedures to provide temporary HVAC during loss of room cooling – Reference SAMA 160

The following additional items were considered based on NRC input:

1. Enhancements to reduce the risk from SG tube rupture – Reference SAMAs 119-123
2. Provisions for alternate power to hydrogen igniters from onsite power sources – Reference SAMA 108
3. Use of existing plant hardware for RCS depressurization – Reference SAMA 42
4. Use of fire water for containment spray or SG makeup – Reference SAMA 75 and 92
5. Use of a hydro pump as a backup for RCP seal injection/cooling – Reference SAMAs 56 and 57

c. *NRC Request*

The basic event CDF importance list in Table 13 only includes four items with a risk reduction worth (RRW) greater than 1.02. Also, no initiating events are included on this list. Discuss the development of this list and why it is limited to only four items.

TVA Response

The WBN plant model has been through numerous revisions and refinements since the original IPE submittal in 1992. Through these revisions, the core damage frequency has, in general, been reduced through model refinement. Also, the contributors to plant risk were subject to further review and refinement such that the number of key contributors to plant risk has been reduced.

Initiating events are not included on the Riskman® importance reports. The following initiating events (see response to Question 1 for detailed listing) contribute 2% or more to core damage frequency and applicable SAMAs are also listed:

SLOCAV (very small LOCA) – SAMA 37
ERCWTL (total loss of ERCW) – SAMAs 155, 220, 271 and 272
CCSTL (total loss of component cooling water) – SAMA 51
CCSA (loss of component cooling train A) – SAMA 51
ERCWB (loss of ERCW train B) – SAMAs 155, 220, 271 and 272
LOSP (loss of offsite power) – SAMAs 11, 12, 13 and 18
SLOCAN (non-isolable small LOCA – RCP seal failure) – SAMAs 37 and 58
FLPH1A (A train flood at intake structure) – SAMAs 155 and 272

d. *NRC Request*

It is stated in Section 6.4 that the systems and basic events that have an RRW of greater than 1.02 for CDF or LERF were reviewed to identify potential SAMAs. For each basic event and system in the importance lists (Tables 11 through 14) indicate the SAMA(s) identified to mitigate the event. For any basic events and systems not addressed by a specific SAMA justify why no such SAMAs were considered.

TVA Response

See listings below for SAMA items related to each of the identified items from tables 11, 12, 13, and 14.

2. Items from Table 1 (System Importance (RRW > 1.02) for CDF)

Rank	System	System Description	Related SAMA
1	ERCW	ESSENTIAL RAW WATER COOLING SYSTEM	155, 220, 271, 272
2	RHR	WBN RESIDUAL HEAT REMOVAL SYSTEM	115, 159
3	CVCS	WBN CHEMICAL AND VOLUME CONTROL SYSTEM	240
4	EPS-AC	AC ELECTRIC POWER SYSTEMS	11, 12, 13
5	CCS	COMPONENT COOLING SYSTEM	49, 50, 51
6	EPS-DC	DC ELECTRIC POWER SYSTEMS	1, 3 ,4, 5, 6
7	RCS	RCS SYSTEMS AND MISC. FUNCTIONS	79, 123
8	AFW	AUXILIARY FEEDWATER SYSTEM	70, 72, 73, 74, 75
9	VENT	VENTILATION SYSTEMS	80, 81, 82, 83

3. Items from Table 2 (System Importance (RRW > 1.02) for LERF)

Rank	System	System Description	Related SAMA
1	ERCW	ESSENTIAL RAW WATER COOLING SYSTEM	155, 220, 271, 272
2	AFW	AUXILIARY FEEDWATER SYSTEM	70, 72, 73, 74, 75
3	EPS-AC	AC ELECTRIC POWER SYSTEMS	11, 12, 13
4	RHR	WBN RESIDUAL HEAT REMOVAL SYSTEM	115, 159
5	AIR	WBN - PLANT COMPRESSED AIR SYSTEMS	86, 87, 89
6	RCS	RCS SYSTEMS AND MISC. FUNCTIONS	79, 123
7	CIS	CONTAINMENT SYSTEMS	93, 94, 97
8	VENT	VENTILATION SYSTEMS	80, 81, 82, 83
9	CVCS	WBN CHEMICAL AND VOLUME CONTROL SYSTEM	240
10	VSEQ	V SEQUENCE EVENTS	111, 117, 118, 182, 239
11	EPS-DC	DC ELECTRIC POWER SYSTEMS	1,3, 4, 5, 6
12	CCS	COMPONENT COOLING SYSTEM	49, 50, 51
13	SEC	SECONDARY SYSTEMS AND FUCNTIONS	66, 71, 76, 77

4. Items from Table 3 (Basic Event Importance (RRW > 1.02) for CDF)

Rank	Basic Event	Basic Event Description	Related SAMA
1	DHARR1	Operators fail to perform alignment for high head recirculation	32,36, 238
2	ERCWGLOBAL	Global Failure of ERCW Pumps	155, 220, 271, 272
3	COVFO1___0620 504	Check valve 62-504 fails to open on demand	273
4	[PMOFR0___0700 0CS PMOFR1___0700 1AA PMOFR1___0700 1BB PMOFR2___0700 2BB]	Common cause failure to run of CCS pumps CS, 1AA, 1BB, 2BB	49, 50, 51

5. Items from Table 4 (Basic Event Importance (RRW > 1.02) for LERF)

Rank	Basic Event	Basic Event Description	Related SAMA
1	ERCWGLOBAL	Global Failure of ERCW Pumps	155, 220, 271, 272
2	FRACT_1AS_NONREC	Fraction 1-AS failures not recoverable	70, 72, 73, 74, 75
3	DHARR1	Operators fail to perform alignment for high head recirculation	32, 36, 238
4	PDMOD23	Relief valve 0-32-512,513, 514,4906,540, or 541 opens prematurely (plant air)	86, 87
5	DHADS2	SGTR with isolation. Steam dumps avail. For cooldown	251
6	FDHADS2	Control flag	42, 77, 203, 235
7	CNTLK1_PREEXISTL	Isolation failure due to large pre-existing leaks	112, 179
8	DHAMU1	Operator failure to open valves 59-737, 738, 511 & 742 and start	33, 67
9	[PTSFS1____00301AS]	Turbine pump 1A-S fails to start on demand	70, 72, 73, 74
10	COVFO1____0620504	Check valve 62-504 fails to open on demand	273
11	[PMSFS1MDP00301AA PMSFS1MDP00301BB PTSFS1____00301AS]	Common cause failure to start of AFW pumps 1AA, 1BB and 1AS	70, 72, 73, 74
12	[1DGS_1AAFS]	DG 1A-A fails to start or run	9, 10
13	[PMOFR0____06700GB]	ERCW pump G-B fails during operation	155, 220, 271, 272
14	[PMOFR0____06700EB]	ERCW pump E-B fails during operation	155, 220, 271, 272
15	[PMOFR0____06700HB]	ERCW pump H-B fails during operation	155, 220, 271, 272
16	[PMOFR0____06700FB]	ERCW pump F-B fails during operation	155, 220, 271, 272
17	DHAOB1	Operator fails to initiate bleed and feed	283

6. Items from Table 5 (Basic Event Importance (RRW > 1.02) for LERF)

Rank	Basic Event	Basic Event Description	Related SAMA
1	ERCWGLOBAL	Global Failure of ERCW Pumps	155, 220, 271, 272
2	FRACT_1AS_NONREC	Fraction 1-AS failures not recoverable	70, 72, 73, 74, 75
3	DHARR1	Operators fail to perform alignment for high head recirculation	32, 36, 238
4	PDMOD23	Relief valve 0-32-512,513, 514,4906,540, or 541 opens prematurely (plant air)	86, 87
5	DHADS2	SGTR with isolation. Steam dumps avail. For cooldown	251
6	FDHADS2	Control flag	42, 77, 203, 235
7	CNTLK1_PREEXISTL	Isolation failure due to large pre-existing leaks	112, 179
8	DHAMU1	Operator failure to open valves 59-737, 738, 511 & 742 and start	33, 67
9	[PTSFS1___00301AS]	Turbine pump 1A-S fails to start on demand	70, 72, 73, 74
10	COVFO1___0620504	Check valve 62-504 fails to open on demand	273
11	[PMSFS1MDP00301AA PMSFS1MDP00301BB PTSFS1___00301AS]	Common cause failure to start of AFW pumps 1AA, 1BB and 1AS	70, 72, 73, 74
12	[1DGS_1AAFS]	DG 1A-A fails to start or run	9, 10
13	[PMOFR0___06700GB]	ERCW pump G-B fails during operation	155, 220, 271, 272
14	[PMOFR0___06700EB]	ERCW pump E-B fails during operation	155, 220, 271, 272
15	[PMOFR0___06700HB]	ERCW pump H-B fails during operation	155, 220, 271, 272
16	[PMOFR0___06700FB]	ERCW pump F-B fails during operation	155, 220, 271, 272
17	DHAOB1	Operator fails to initiate bleed and feed	283

e. <u>NRC Request</u>

The Unit 1 IPEEE (Table 3.1.4-1) identifies the high confidence low probability of failure (HCLPF) values for a number of components which could not be screened *out. For those items with HCLPF values below approximately 0.7g (the value corresponding to a seismic CDF approximately equal to the internal events CDF) identify potential SAMAs that might address the limiting failure mode, and justify why these SAMAs should not be considered further. In assessing the feasibility of implementing a SAMA at Unit 2 include consideration of the status of construction or modification of the affected structures or components at Unit 2.*

TVA Response

The components identified in Table 3.1.4-1 are listed in the table below. SAMAs associated with seismic issues are also listed below. Of the identified SAMAs, two were screened based on excessive cost, whereas the other four were screened based on the IPEEE guideline of 0.30g force.

ITEM	BLDG.	RESOLUTION * HCLPF	ISSUE	NOTES
Spatial Interactions	Various	Interactions determined to be acceptable	Seismic Spatial Interaction	WCG-1-1842
Masonry Walls	Various	0.53g	Stability	WCG-1-1843
6900V Shutdown Boards	AUXILIARY	0.45g	Structural Integrity	WCG-1-1846
DG Air Intake Filters	D.G.	1.78g	Anchorage	WCG-1-1847
Main Control Rm AHU	CONTROL	0.56g	Anchorage	WCG-1-1848
Auxiliary Bldg. Roof Diaphragm	AUXILIARY	0.75g	Structural Integrity	WCG-1-1850
RHR Pumps	AUXILIARY	0.50g	Anchorage	WCG-1-1851
480V Shutdown Board Transformers	AUXILIARY	0.38g	Structural Integrity	WCG-1-1852
Control Air Pre ~ Aft. Filters	AUXILIARY	1.08g	Anchorage	WCG-1-1853
CCS Heat Exchangers	AUXILIARY	0.38g	Anchorage	WCG-1-1854
ERCW Pumps	IPS	>0.40g	Anchorage	WCG-1-1855
IPS Screen Wash Pumps	IPS	0.36g	Anchorage	WCG-1-1855
480V -Reactor MOV, Reactor Vent, Control & Aux. Boards	AUXILIARY	0.40g	Anchorage	WCG-1-1856

ITEM	BLDG.	RESOLUTION *HCLPF	ISSUE	NOTES
Main Cont. Rm: Ceiling Structure *	CONTROL	0.36g *	Anchorage	WCG-1-1857 *
Main Cont. Rm: Electrical Panels	CONTROL	0.70g	Structural Integrity	WCG-1-1857

* Note: Changes subsequent to the Unit 1 IPEEE Report to the Main Control Room Ceiling have increased its seismic ruggedness. Calculation WCG-2-617 performed for the Unit 2 IPEEE Report shows the HCLPF value for the ceiling structure is now 0.52g.

The following SAMAs address these items with HCLPF values below approximately 0.7g.

- SAMA 99 – Strengthen primary/secondary containment (e.g., add ribbing to containment shell) – Due to plant construction status, cost of implementation would exceed bounding benefit.
- SAMA 140 – Increase seismic ruggedness of plant components – screened based on seismic screening value of 0.3 g from IPEEE walkdowns.
- SAMA 177 – Replace anchor bolts on EDG oil cooler – screened – not identified on seismic margin review for IPEEE.
- SAMA 212 – Improve seismic capacity of walls near 4160/600 VAC transformers – screened based on seismic margin review.
- SAMA 213 – Improve seismic capacity of the EDG fuel oil day tanks – screened based on seismic margin review.
- SAMA 214 – Reinforce the seismic capacity of the steel structure supporting the auxiliary building – screened based on excessive cost, given construction status of Aux Bldg.

f. <u>*NRC Request*</u>

The Unit 1 IPEEE fire PRA identified a number of fire-initiated core damage sequences with a CDF greater than 3E-07 per year (the equivalent of the RRW screening criteria of 1.02 used in the selection of SAMA candidates from the Unit 2 internal events PRA). For each of these fire sequences, identify potential SAMAs that might reduce the fire risk either individually or as a group, and justify why these SAMAs should not be considered further.

<u>TVA Response</u>

The following fire-related CDF greater than 3E-07 scenarios were taken from the detailed analysis described in Attachment 4 of TVA's response to Generic Letter 88-20, Supplements 4 and 5 - Individual Plant Examination of External Events (IPEEE) For Severe Accident Vulnerabilities (RIMS T04 980217 539) :

Room 708.0-C1 (Unit 1 Auxiliary Instrument Room)
Room 708.0-C4 (Unit 2 Auxiliary Instrument Room)
Room 755.0-C12 (Main Control Room)
Room 713.0-A1 (Corridor)
Room 757.0-A2 (6.9kV and 480V Shutdown Board Room A)
Room 757.0-A5 (480V Shutdown Board Room 1B)
Room 757.0-A24 (6.9kV and 480V Shutdown Board Room B)
Room 772.0-A4 (125V Vital Battery Room I)
Room 772.0-A14 (125V Vital Battery Room III)
Room 772.0-A15 (480V Board Room 2B)

SAMAs for internal fire events are described in Section 6.3.2 of the SAMA report. The following SAMAs address the fire related scenarios above.

SAMA 142 (replace mercury switches in fire protection system)
SAMA 143 (upgrade fire compartment barriers)
SAMA 144 (install additional transfer and isolation switches)
SAMA 145 (enhance fire brigade awareness)
SAMA 146 (enhance control of combustibles and ignition sources)
SAMA 256 (install fire barriers around cables or re-route cables away from fire source)

Of these SAMAs, four (SAMAs 142, 144, 145 and 146) were screened as already implemented, SAMA 143 was screened due to excessive cost, and SAMA 256 was retained for Phase II evaluation.

g. <u>*NRC Request*</u>

The description of the Phase I screening criteria in Section 7 implies that in order for an item to be screened out as "already implemented" it must be implemented and accounted for in the PRA model. The majority of the SAMA candidates identified through the RRW review (listed in Table 15) were screened out as "already implemented." If these SAMA candidates were accounted for in the PRA, then the failure they address must still be important enough to have been identified in the RRW review. For each of the items in Table 15 screened out as "already implemented" (i.e., SAMAs 3, 12, 75,1 57, 198, 244, 257, 271,272,275), identify additional SAMA candidates that would address the failure and provide a further evaluation of these SAMAs.

<u>TVA Response</u>

The specific actions may be either implemented (maintenance cross-ties for 6.-9kV shutdown boards) in the plant or in the implementation process (see discussion of SAMA 275 – design change in to model in progress – under response to RAI 5.h.iii below). Data related items whose purpose is to improve reliability (SAMA 198 – improve RHR sump reliability) will be captured in the PRA through actual plant data history collected through the PRA update process. In each of these cases, while the intent is to update the PRA model to reflect these changes, they are not yet accounted for in the SAMA plant model.

- SAMA 3 (add additional battery charger or portable, diesel driven charger to existing system) – Two spare battery chargers are installed at WBN. The spare chargers are modeled in the PRA model. The intent of this SAMA is to provide for improved availability of the DC power system. To improve DC power system availability, a spare battery (Vital Battery #5) is also installed at WBN. The additional battery is accounted for in the PRA by not modeling maintenance of the four vital batteries.

- SAMA 12 (create AC power cross-tie capability with other unit) – WBN has procedures that address the capability of being able to cross-tie power at the 6.9-kV shutdown board level. This is not yet accounted for in the PRA model used in the SAMA analysis.

- SAMA 75 (use fire water system as a backup for SG inventory) – Using the high pressure fire protection (HPFP) system to provide inventory to the SGs is procedurally implemented at WBN in response to a site flooding event; however, this use of the HPFP system is not modeled in the SAMA PRA.

- SAMA 157 (implement procedure guidance for use of cross-tied CCW or SW [i.e., ERCW] pumps) – A recent hardware modification, including procedural guidance, was made to the ERCW system at WBN to allow ERCW headers to be cross-tied. This is not credited in the SAMA PRA model.

- SAMA 198 (improve RHR sump reliability) - Modifications were made to the containment sump strainers in response to GSI-191. These modifications do not impact the PRA model. The benefits of this modification will be realized in the PRA through reduced failure probabilities and be implemented through data updates of the PRA. An update of this data was not performed in the PRA used for the SAMA analysis.

- SAMA 244 (AC cross-tie capability) - WBN has procedures that address the capability of being able to cross-tie power at the 6.9-kV shutdown board level. This is not yet accounted for in the PRA model used in the SAMA analysis.

- SAMA 257 (inter-train CCW cross-tie for emergency operation) - The plant is capable of cross tying CCS trains, but this is not credited in the SAMA PRA.

- SAMA 271 (refurbish ERCW pumps and upgrade capacity of current pumps) - WBN is currently replacing ERCW pumps to support two-unit operation. This modification does not change the PRA as modeled and will be implemented into the model in the future through data updates of pump reliability and availability.

- SAMA 272 (provide portable diesel powered 5,000 gpm pump as backup to ERCW system) - At WBN, a diesel-driven HPFP pump is available to supply ERCW headers if needed. This feature is not modeled in the SAMA PRA.

- SAMA 275 (provide a new inverter arrangement) – A modification was implemented at WBN providing additional inverters. This modification was installed after the latest freeze date of the PRA model and therefore not included in the SAMA PRA model used in this analysis.

h. <u>*NRC Request*</u>

Provide further information on the basis for the disposition of the following SAMAs in Table 16:

i. *SAMA 3, provide a diesel-driven battery charger, is considered already implemented on the basis that two spare chargers are available. Spare chargers are not equivalent to a portable diesel driven charger. Further justify the screening of this SAMA, or provide a Phase II evaluation.*

<u>TVA Response</u>

SAMA 3 was taken from NEI 05-01 that discussed the use of spare charger capability OR portable, diesel-driven battery charger. The first of the two options cited in the SAMA was implemented in the plant design.

ii. <u>*NRC Request*</u>

SAMA 5, provide DC bus cross-ties, is considered to have very little benefit due to existing cross-tie capabilities. This conclusion is not obvious without additional information on the potential benefits and costs. Also, this system/SAMA does not meet the criteria for screening based on very low benefit (which is that it is a nonrisk significant system). Further justify the screening of this SAMA, or provide a Phase II evaluation.

<u>TVA Response</u>

The purpose of this SAMA was to provide for increased availability of the DC power system. The current WBN design allows for DC bus cross-tie at the 480-VAC bus level; also, the system is designed with a spare #5 vital battery that can be aligned to supply any of the 4 DC buses. The cross-ties are not credited in the model used for the SAMA analysis, and the spare vital battery is accounted for by not modeling maintenance for the other four batteries. Beyond this level of flexibility, the addition of even more cross-tie capability was seen to add only marginal improvement to system reliability.

iii. *NRC Request*

For a number of SAMAs (e.g., SAMAs 16, 65, 275 and 281), it is indicated that design changes are in process or actions will be taken. Confirm that TVA is committed to these design changes and actions, and describe these commitments.

TVA Response

Regarding the status of the following SAMA related design changes:

- SAMA 16 (improve uninterruptible power supplies) – the new inverters have been installed.
- SAMA 65 (install digital feedwater upgrade) – the design change is in process and will be implemented for Unit 2 prior to fuel load.
- SAMA 174 (replace batteries) – shared vital batteries I and II were replaced within the past 2 years; shared vital batteries III and IV will be replaced in a timeframe between 2012 to 2015.
- SAMA 188 (implement modifications to the compressed air system [Unit 1 control air compressor] to increase the capacity of the system) – A construction compressor has been linked into the Unit 1 system to increase the capacity of the system.
- SAMA 198 (improve RHR sump reliability) – the design change has been issued and will be implemented for Unit 2 prior to fuel load.
- SAMA 218 (improve reliability of power supplies) – the design changes to accomplish this SAMA will be complete by the end of this year except as noted for the batteries mentioned above in SAMA 174.
- SAMA 219 (improve switchyard and transformer reliability) – the design changes to accomplish this item will be completed by the end of the year.
- SAMA 269 (provide two trains of cooling to the 480-V board room) – two trains cooling in the 480-V board room exists. Ventilation for Unit 1 is cooled by trains 1A and 1B; ventilation for Unit 2 is provided by 2A and 2B.
- SAMA 271 (refurbish the ERCW pumps and upgrade the capacity of the current pumps) – the design change has been issued and will be implemented for Unit 2 prior to fuel load.
- SAMA 275 (provide a new inverter arrangement) – the new inverters have been installed.
- SAMA 281 (replace ACAS compressors and dryers) – WBN will refurbish and enhance current units to gain capacity, but not replace units.

iv. *NRC Request*

SAMA 19, use fire water system as backup source for diesel cooling, was screened on the basis that the opposite train of emergency raw cooling water (ERCW) is available as a backup. Since the ERCW provides both the primary *and backup cooling and is subject to common cause failures, it is not equivalent to using the fire water system, which adds a diverse method of cooling. Further justify the screening of this SAMA, or provide a Phase II evaluation.*

TVA Response

In the event of failure of ERCW, a diesel-driven fire pump is available to supply water to ERCW headers.

v. *NRC Request*

SAMA 29, provide capability for alternate injection via diesel-driven fire pump, is considered to be of minimal benefit since it does not provide a recirculation path and is considered cost-prohibitive relative to the potential benefit. This conclusion is not obvious without additional information on the potential benefits and costs. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

In addition to not providing a recirculation path (i.e., single pass-through cooling, rather than a closed system), the use of fire protection would require plant depressurization to less than 150 psig, requiring an excessive inventory of makeup. This is seen as being of marginal benefit relative to the cost of design and installation. Also, fire protection makeup to the SGs is procedurally addressed for the case of site flooding situations.

vi. *NRC Request*

SAMA 42, procedure change for reactor coolant system depressurization, was not subjected to a cost-benefit analysis since emergency operating procedures are processed through the owner's group emergency response guideline (ERG) maintenance process. This is not a valid reason for screening. If procedure change is cost beneficial and desirable, it should be pursued through the ERG maintenance process. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

TVA has reviewed its ERGs and SAMG procedures. TVA procedure SAG-2, "Depressurize the RCS," contains guidance to depressurize the RCA and allow RCS makeup from low pressure injection sources. TVA now considers SAMA 42 already implemented.

vii. *NRC Request*

SAMA 48, cap downstream piping of normally closed component cooling water (CCW) train and vent valves, is considered already implemented but the comment does not indicate that caps are in place. Rather, it states that failure of the drain or vent valve to remain closed is less likely than failure of the socket weld connection itself. Provide the basis for the latter statement.

TVA Response

From NUREG/CR-6928, a small (less than 50 gpm) leak through manual valves is given an hourly failure rate of 7.0E-8. This gives an annual failure rate per vent/drain valve section of:

7.0E-8 x 8760 hours/year = 6.13E-4 valve leak failures/year per vent/drain valve

While the welded sections of pipe are shown with an hourly failure rate (again less than 50 gpm) of 2.5E-10, for operation of the vent/drain, the end cap will need to be removable (i.e., threaded), such that failure following re-installation is judged to be somewhat higher than the failure rate of the seated valve itself.

viii. *NRC Request*

SAMA 53, shed CCW loads upon loss of ERCW, is considered to have little benefit since this affects recovery of ERCW and the PRA takes no credit for ERCW recovery. If load shedding can improve the probability of recovering ERCW then the benefit might be important, particularly since loss of ERCW is an important contributor to risk. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

As stated in Phase I Comment for this SAMA, while Loss of ERCW AOI-13 does not direct the operator to Loss of CCS AOI-15, it does direct the operator to trip all RCPs, isolate thermal barrier cooling, cool down the plant, and cross-tie ERCW to the opposite train, if available.

While ERCW recovery is not specifically modeled in the PRA, given the level of actions currently provided in AOI-13, directing the operator to also concurrently perform CCS load shed is judged to be of marginal additional benefit when the emphasis should be on recovery of the ERCW train including connecting the diesel-driven HPFP pump to ERCW.

ix. <u>NRC Request</u>

SAMA 54, increase charging pump lube oil capacity, is considered already implemented on the basis that alternate cooling is available to the WBN A charging pump. The SAMA would benefit the A pump even with the added cooling and the B charging pump that apparently does not have the added cooling. Further justify the screening of this SAMA, or provide a Phase II evaluation.

<u>TVA Response</u>

This SAMA is intended to address changes that could be made before charging pump failure due to lube oil overheating during loss of cooling water sequences. At WBN, the focus was to provide alternate cooling to the charging pump rather than increasing the lube oil capacity. Based on input from Westinghouse engineering, the heatup rate for a larger lube oil sump is not linear, such that doubling sump capacity would not double the time available prior to CCP failure so that performing the action to provide alternate cooling would still be necessary. In the HRA supporting the SAMA model (based on the WBN Rev 4 model), the operator action to align ERCW to cool the charging pump bearing and gear box lube oil coolers, HCCSR2, has a failure rate of 2.7E-02. This is based on a 5-minute period to diagnose and perform the action, including transit to the charging pump area. The time to irreversible consequences is based on the single pump operation without cooling (10 minutes). Within the HRA calculator, extending Tsw to 15 minutes did not change the operator error rate, indicating that the operator currently has adequate time to recover, based on the time to perform the action and the time before irreversible consequences.

Also, it should be noted that the 10-minute value used for Tsw assumes the time to fail one charging pump, whereas the operator would secure the A charging pump (if running) and start the B pump, allowing it to run to failure, if necessary, while aligning ERCW, such that the proposed SAMA would not improve the operator reliability from the currently analyzed case.

x. <u>NRC Request</u>

SAMA 58, install improved reactor coolant pump (RCP) seals, was not subjected to a cost-benefit analysis since the cost for a new design by Westinghouse is not available and this SAMA is not under TVA control. Southern Nuclear Company estimated the cost of installation of an improved RCP seal for Westinghouse reactors to be about $1M for the Vogtle Electric Generating Plant (VEGP) (see VEGP Units 1 and 2 License Renewal Application Environmental Report). Considering this cost estimate, further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

SAMA 56 for improved RCPs seals can be compared to SAMA 156, which received a Phase II analysis. SAMA 156 has an estimated benefit of RCP seal improvement as $675,053. This gives a benefit/cost ratio of 0.675, when compared with the $1 million VEGP cost estimate cited above. Also, TVA has received an estimate for the improved seals from Westinghouse at $2.2 million to install on both units (in other words, $1.1 million each). This reduces the benefit/cost ratio to 0.614. Adjusting for the 95% CDF and LERF, as shown in uncertainty analysis Table 19, increases the ratio to 1.71, such that, by this consideration alone, the SAMA could be considered potentially cost-beneficial. As noted in Section 9.2 of the SAMA report, this is only one of the considerations used in the decision process. Accounting for all of the applicable considerations, the cost-benefit ratio for this SAMA shown in Table 20 would then be consistent across all columns at 0.6.

xi. *NRC Request*

SAMA 64, manually align the fire water system to the CCW system, is considered already implemented on the basis that ERCW is available to provide cooling to the residual heat removal system. It would appear that use of the fire water system would provide additional benefit for loss of ERCW. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

The intent of this SAMA is to improve the ability to cool the RHR heat exchangers on a loss of CCW. Use of ERCW rather than fire water as supplemental RHR heat exchanger cooling is a viable option for this initiating event, unless there is a concurrent loss of the normally assigned train of 4 (2 normally running, 2 standby) ERCW pumps for cooling one train of ERCW cooling. This is judged to be an unlikely conditional failure, such that the proposed SAMA would be of marginal incremental benefit.

xii. *NRC Request*

SAMA 74, provide hookup for portable generators to power the turbine-driven auxiliary feedwater (AFW) pump after battery depletion, is considered already implemented on the basis of an extra battery and an alternate power supply for the battery charger are already available. An independent source of power to control the turbine-driven AFW pump would appear to have a benefit beyond that provided for at WBN. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

The 5th vital battery acts as a full spare, such that it would provide a replacement following depletion of normal DC power provided by the other four batteries. Also, procedures have been developed to allow operation of the turbine driven AFW pump following battery depletion, meeting the intent of the SAMA.

xiii. *NRC Request*

SAMA 80, provide a redundant train or means of ventilation, is considered to have very low benefit based on current provisions for compensatory ventilation and plant modifications are in progress. However, the ventilation system has a RRW of > 1.02, and does not meet the criteria for screening based on very low benefit (which is that it is a nonrisk significant system.). Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

As noted in the Phase I Comments, plant chillers are being upgraded based on Freon considerations. This will also improve the reliability of the units, which have had a history of problematic operation. Also, heatup calculations are being updated, and that has the potential to remove ventilation as a dependency in many areas, given the extremely long room heatup times. This evaluation is extremely conservative in that the dependency is assumed to be catastrophic and fast acting, whereas actual room and area heatup occurs over several hours, if not several days, allowing for the addition of temporary ventilation to the areas. Given these considerations, the intent of the SAMA is judged to be adequately addressed.

xiv. *NRC Request*

SAMA 111, install additional pressure or leak monitoring instruments for detection of interfacing system loss-of-coolant accidents (ISLOCAs), is considered already implemented on the basis that instrumentation and procedures for responding to an ISLOCA are in place. However, the intent of this SAMA is to add instruments that would give a warning of a potential ISLOCA. Further justify the screening of this SAMA, or provide a Phase II evaluation.

TVA Response

ISLOCA initiators currently contribute 0.09% of CDF (see results for initiating events XI and XS in response to RAI 1.a). This gives a maximum benefit of 0.0009 x $1,535,812 = $1,382 for this SAMA.

xv. *NRC Request*

SAMA 262, provide connections from the "B" centrifugal charging pump to the ERCW system, is considered to have very low benefit based on an evaluation. In Section 6.2.2, a CDF decrease of 4 percent is given for this SAMA. The disposition of this SAMA is not in accord with the criteria for screening based on a very low benefit, which requires the item to involve a nonrisk significant system. Further, this SAMA is listed in Table 15 as having been identified from the RRW review. Provide the results of a Phase II evaluation for this SAMA.

TVA Response

This change was evaluated in the 1994 SAMDA report (see item IV.3 under RAI 5.b above) with a cost of $295,200 in 1994 dollars. Adjusting for inflation at an average of 3% per year gives an inflation-adjusted cost for 15 years of 1.56, for a current cost of $460,512. The corresponding benefit for this change is associated with a 4% reduction in CDF = 4% x $1,535,812 bounding benefit = $61,432. This gives a cost-to-benefit ratio of 7.5, such that this change is not cost effective.

6. *NRC Request*

Provide the following information with regard to the Phase II cost-benefit evaluations:

a. *Provide a brief description of the process used to develop the cost estimates for implementing the Phase II SAMAs. Identify the cost factors that are included in the cost estimates. Clarify whether the cost estimates include: lifetime testing and maintenance costs, contingency costs associated with unforeseen implementation obstacles, or inflation.*

TVA Response

SAMA cost estimates are focused on labor (craft, engineering, etc.) and component costs related to installation of the proposed physical change. Costs do not include lifetime operation, testing or maintenance costs or contingency related to unforeseen obstacles or inflation. These considerations would drive the cost higher, making the screening more likely, such that not including them will make the SAMA more attractive and likely to be retained from a cost/benefit perspective.

b. *NRC Request*

Provide the percent change in the population dose risk and offsite economic cost risk for each Phase II SAMA so that the benefits presented can be confirmed.

TVA Response

See the two tables below. Table 1 provides the calculation of the base case risk and potential (bounding) cost while Table 2 shows the pertinent information in column format for each of the Phase II SAMAs.

Table 1

OUTPUTS						
Release Category	Pop Dose (Sv)	**Econ Cost ($)**	**Pop Dose (rem)**	**RC Frequencies per year**	**Pop Dose (man-rem/year)**	**Econ Cost $/year**
I	2.19E+04	**4.45E+09**	**2.19E+06**	2.6379E-07	5.78E-01	1.17E+03
II	3.42E+04	**8.11E+09**	**3.42E+06**	1.1919E-07	4.08E-01	9.67E+02
III	1.16E+04	**1.78E+09**	**1.16E+06**	1.9950E-06	2.31E+00	3.55E+03
				1.2990E-05		
Total				1.54E-05	3.30E+00	5.69E+03

Off-Site Exposure Cost

Wpha	**$88,541**	**off-site exposure cost ($)**
C	13.41699911	[1-exp(-rtf)]/r (years)
Tf	40	analysis period (years)
R	0.07	real discount rate (7% = 0.07/year)
Zpha	6.60E+03	value of public health (accident) risk per year before discounting ($/year)
Zpha	$2000/person-rem	* mean annual off-site dose impact due to a severe accident

Off-Site Economic Cost

Wpha	**$76,365**	**off-site economic cost ($)**
C	13.41699911	[1-exp(-rtf)]/r (years)
Tf	40	analysis period (years)
R	0.07	real discount rate (7% = 0.07/year)
Zea	$5,692	mean annual economic impact due to a severe accident

On-Site Exposure Cost - Immediate Dose

WIO	$1,361	immediate on-site exposure cost ($)
R	$2,000	monetary equivalent of unit dose ($/person-rem)
F	1.54E-05	Level 1 internal events core damage frequency (events/year)
DIO	3300	immediate on-site (occupational) dose (person-rem/event)
C	13.41699911	[1-exp(-rtf)]/r (years)
R	0.07	real discount rate (7% = 0.07/year)
Tf	40	analysis period (years)

On-Site Exposure Cost - Long-Term Dose

WLTO	$5,931	long-term on-site exposure cost ($)
R	$2,000	monetary equivalent of unit dose ($/person-rem)
F	1.54E-05	Level 1 internal events core damage frequency (events/year)
DLTO	20000	long-term on-site (occupational) dose (person-rem/event)
C	13.41699911	[1-exp(-rtf)]/r (years)
R	0.07	real discount rate (7% = 0.07/year)
Tf	40	analysis period (years)
M	10	years over which long-term doses accrue

Total On-Site Exposure

WO	**$7,292**	**(WIO + WLTO)**

On-Site Economic Cost - Cleanup and Decontamination

PVCD	$1,078,745,778	net present value of a single event ($)
CCD	$1,500,000,000	total cost of cleanup and decontamination effort ($)
M	10	cleanup period (years)
R	0.07	real discount rate (7% = 0.07/year)

UCD	$14,473,531,133	total cost of cleanup and decontamination over the analysis period ($-years)
PVCD	$1,078,745,778	net present value of a single event ($)
C	13.41699911	[1-exp(-rtf)]/r (years)
R	0.07	real discount rate (7% = 0.07/year)
Tf	40	analysis period (years)

On-Site Economic Cost - Replacement Power Cost

PVRP	$1,927,554,493	net present value of replacement power for a single event ($)
R	0.07	real discount rate (7% = 0.07/year)
Tf	40	analysis period (years)
B	$152,967,033	a constant representing a string of replacement power costs that occur over the lifetime of a reactor after an event (for a 910 MWe "generic" reactor, NUREG/BR-0184 uses a value of $1.2E+08) ($/year)
	1160	Watts Bar Mwe

URP	$24,289,327,255	net present value of replacement power over life of facility ($-year)
R	0.07	real discount rate (7% = 0.07/year)
Tf	40	analysis period (years)

On-Site Economic Cost includes cleanup and decontamination cost, and either replacement power cost or repair and refurbishment cost.

Total On-Site Economic Costs

Total	$595,708	
CDF	1.537E-05	interval events CDF

Total Cost of Severe Accident Risk/Maximum Benefit	
Sum of the baseline costs:	
Off-site exposure cost	$88,541
Off-site economic cost	$76,365
On-site exposure cost	$7,292
On-site economic cost	$595,708
Severe Accident Impact	**$767,906**
Combine the severe accident impact with the external events multiplier to calculate the total cost of severe accident risk.	
Severe Accident Impact	$767,906
External Events Multiplier	2
Maximum Benefit	**$1,535,812**

Table 2

	Base	SAMA 4	SAMA 8	SAMA 32	SAMA 45	SAMA 46
RISKMAN Output						
RC 1 Frequency (per yr)	2.6379E-07	2.3821E-07	2.55316E-07	2.16597E-07	2.56035E-07	2.4024E-07
RC 2 Frequency (per yr)	1.1919E-07	1.1374E-07	1.18904E-07	9.51989E-08	1.14645E-07	1.1460E-07
RC 3 Frequency (per yr)	1.9950E-06	1.6230E-06	1.96858E-06	1.67192E-06	1.96539E-06	1.9068E-06
RC 4 Frequency (per yr)	1.2990E-05	1.2957E-05	1.28158E-05	7.3453E-06	1.1986E-05	1.2025E-05
CDF Total	1.5368E-05	1.4932E-05	1.5159E-05	9.3290E-06	1.4322E-05	1.4287E-05
RC 1 Pop dose (man-rem/yr)	5.7770E-01	5.2167E-01	5.5914E-01	4.7435E-01	5.6072E-01	5.2614E-01
RC 2 Pop dose (man-rem/yr)	4.0765E-01	3.8899E-01	4.0665E-01	3.2558E-01	3.9209E-01	3.9192E-01
RC 3 Pop dose (man-rem/yr)	2.3142E+00	1.8827E+00	2.2836E+00	1.9394E+00	2.2798E+00	2.2118E+00
Total Pop dose (man-rem/yr)	3.2996E+00	2.7933E+00	3.2493E+00	2.7394E+00	3.2327E+00	3.1299E+00
RC 1 Econ Cost ($/yr)	1.1739E+03	1.0600E+03	1.1362E+03	9.6385E+02	1.1394E+03	1.0691E+03
RC 2 Econ Cost ($/yr)	9.6667E+02	9.2244E+02	9.6431E+02	7.7206E+02	9.2977E+02	9.2937E+02
RC 3 Econ Cost ($/yr)	3.5511E+03	2.8889E+03	3.5041E+03	2.9760E+03	3.4984E+03	3.3940E+03
Total Econ Cost ($/yr)	5.6917E+03	4.8714E+03	5.6045E+03	4.7119E+03	5.5675E+03	5.3925E+03
Off-Site Exposure Cost						
Zpha	$6,599	$ 5,587	$6,499	$5,479	$6,465	$6,260
Wpha	$88,541	$74,957	$87,193	$73,508	$86,745	$83,987
Off-Site Economic Cost						
Zea	$5,692	$4,871	$ 5,605	$ 4,712	$5,568	$5,392
Wpha	$76,365	$ 65,360	$75,196	$63,220	$ 74,699	$72,351
On-Site Exposure Cost						
WIO	$ 1,361	$1,322	$ 1,342	$826	$ 1,268	$ 1,265
WLTO	$ 5,931	$5,763	$5,851	$ 3,601	$ 5,528	$5,514
Total	$7,292	$ 7,085	$7,193	$4,427	$ 6,796	$ 6,779

	Base	SAMA 4	SAMA 8	SAMA 32	SAMA 45	SAMA 46
RISKMAN Output						
On-Site Economic Costs						
Cleanup and Decontamination						
PVCD	$1,078,745,778	$1,078,745,778	$1,078,745,778	$1,078,745,778	$ 1,078,745,778	$1,078,745,778
UCD ($-yr)	$14,473,531,133	$ 14,473,531,133	$14,473,531,133	$14,473,531,133	$14,473,531,133	$14,473,531,133
Replacement Power Cost						
PVRP	$1,927,554,493	$1,927,554,493	$1,927,554,493	$ 1,927,554,493	$1,927,554,493	$ 1,927,554,493
URP ($-yr)	$24,289,327,255	$24,289,327,255	$ 24,289,327,255	$ 24,289,327,255	$ 24,289,327,255	$24,289,327,255
Total On-Site Economic Costs ($)	$ 595,708	$ 578,805	$ 587,590	$ 361,619	$ 555,164	$553,789
Total Cost of Severe Accident Risk	$ 767,906	$ 726,207	$ 757,172	$ 502,774	$ 723,404	$ 716,906
External Event Multiplier (2)	1,535,812.07	1,452,413.13	1,514,343.07	1,005,547.86	1,446,808.99	1,433,812.55
Change from Base		$ (83,399)	$ (21,469)	$ (530,264)	$ (89,003)	$ (102,000)
Costs (from TVA)		$31,675	$26,773	$ 2,100,000	$31,675	$1,042,511
CB Ratio		-2.63	-0.80	-0.25	-2.81	-0.10
RISKMAN Output						
RC 1 Frequency (per yr)	2.03848E-07	2.6555E-07	5.2366E-06	1.7195E-07	2.4932E-07	2.6122E-07
RC 2 Frequency (per yr)	9.02165E-08	1.1936E-07	1.7088E-07	8.9231E-08	1.1919E-07	1.1891E-07
RC 3 Frequency (per yr)	1.41957E-06	2.0016E-06	2.2575E-05	1.5631E-06	1.9950E-06	1.9912E-06
RC 4 Frequency (per yr)	6.1881E-06	1.2991E-05	9.0943E-06	1.3032E-05	1.2990E-05	1.2917E-05
CDF Total	7.9017E-06	1.5377E-05	3.7077E-05	1.4856E-05	1.5354E-05	1.5288E-05
RC 1 Pop dose (man-rem/yr)	4.4643E-01	5.8155E-01	1.1468E+01	3.7656E-01	5.4601E-01	5.7207E-01
RC 2 Pop dose (man-rem/yr)	3.0854E-01	4.0820E-01	5.8441E-01	3.0517E-01	4.0765E-01	4.0669E-01
RC 3 Pop dose (man-rem/yr)	1.6467E+00	2.3218E+00	2.6187E+01	1.8132E+00	2.3142E+00	2.3098E+00
Total Pop dose (man-rem/yr)	2.4017E+00	3.3116E+00	3.8240E+01	2.4949E+00	3.2679E+00	3.2885E+00
RC 1 Econ Cost ($/yr)	9.0713E+02	1.1817E+03	2.3303E+04	7.6516E+02	1.1095E+03	1.1624E+03
RC 2 Econ Cost ($/yr)	7.3166E+02	9.6799E+02	1.3858E+03	7.2366E+02	9.6667E+02	9.6440E+02
RC 3 Econ Cost ($/yr)	2.5268E+03	3.5628E+03	4.0184E+04	2.7823E+03	3.5511E+03	3.5443E+03
Total Econ Cost ($/yr)	4.1656E+03	5.7125E+03	6.4872E+04	4.2711E+03	5.6273E+03	5.6711E+03
Off-Site Exposure Cost						
Zpha	$4,803	$6,623	$76,479	$4,990	$6,536	$6,577
Wpha	$ 64,446	$88,863	$1,026,123	$66,949	$ 87,690	$ 88,245

	SAMA 56	SAMA 70	SAMA 71	SAMA 87	SAMA 112	SAMA 136
RISKMAN Output						
Off-Site Economic Cost						
Zea	$ 4,166	$ 5,712	$64,872	$4,271	$ 5,627	$ 5,671
Wpha	$55,890	$ 76,644	$870,392	$ 57,306	$75,501	$ 76,090
On-Site Exposure Cost						
WIO	$ 700	$1,362	$3,283	$1,316	$ 1,360	$ 1,354
WLTO	$ 3,050	$ 5,935	$14,310	$5,734	$ 5,926	$ 5,901
Total	$ 3,749	$ 7,297	$ 17,593	$ 7,049	$ 7,285	$ 7,255
On-Site Economic Costs						
Cleanup and Decontamination						
PVCD	$1,078,745,778	$1,078,745,778	$1,078,745,778	$ 1,078,745,778	$1,078,745,778	$1,078,745,778
UCD ($-yr)	$ 14,473,531,133	$ 14,473,531,133	$ 14,473,531,133	$ 14,473,531,133	$ 14,473,531,133	$14,473,531,133
Replacement Power Cost						
PVRP	$ 1,927,554,493	$ 1,927,554,493	$ 1,927,554,493	$1,927,554,493	$ 1,927,554,493	$1,927,554,493
URP ($-yr)	$24,289,327,255	$24,289,327,255	$ 24,289,327,255	$24,289,327,255	$ 24,289,327,255	$ 24,289,327,255
Total On-Site Economic Costs ($)	$ 306,294	$596,075	$1,437,205	$575,872	$ 595,147	$592,619
Total Cost of Severe Accident Risk	$430,379	$768,879	$3,351,313	$707,176	$ 765,624	$764,208
External Event Multiplier (2)	860,758.96	1,537,757.14	6,702,625.66	1,414,351.75	1,531,247.33	1,528,415.42
Change from Base	$(675,053)	$1,945	-	$(121,460)	$(4,565)	$ (7,397)
Costs (from TVA)	$2,400,000	$ 256,204	$ 1,706,586	$ 886,205	$ 691,524	$ 241,795
CB Ratio	-0.28	0.01	0.00	-0.14	-0.01	-0.03
RC 1 Frequency (per yr)	2.03848E-07	2.4978E-07	2.3821E-07	1.9784E-07	2.5597E-07	2.6307E-07
RC 2 Frequency (per yr)	9.02165E-08	1.1854E-07	1.1374E-07	1.0237E-09	1.1528E-07	1.1528E-07
RC 3 Frequency (per yr)	1.41957E-06	1.8147E-06	1.6230E-06	1.4963E-06	1.9392E-06	1.9896E-06
RC 4 Frequency (per yr)	6.1881E-06	1.2945E-05	1.2957E-05	9.7425E-06	1.2066E-05	1.2954E-05
CDF Total	7.9017E-06	1.5128E-05	1.4932E-05	1.1438E-05	1.4376E-05	1.5322E-05
RC 1 Pop dose (man-rem/yr)	4.4643E-01	5.4703E-01	5.2167E-01	4.3328E-01	5.6057E-01	5.7612E-01
RC 2 Pop dose (man-rem/yr)	3.0854E-01	4.0541E-01	3.8899E-01	3.5009E-03	3.9427E-01	3.9427E-01
RC 3 Pop dose (man-rem/yr)	1.6467E+00	2.1051E+00	1.8827E+00	1.7357E+00	2.2494E+00	2.3079E+00

	SAMA 156	SAMA 176	SAMA 243	SAMA 256	SAMA 273	SAMA 276
RISKMAN Output						
Total Pop dose (man-rem/yr)	2.4017E+00	3.0575E+00	2.7933E+00	2.1724E+00	3.2043E+00	3.2783E+00
RC 1 Econ Cost ($/yr)	9.0713E+02	1.1115E+03	1.0600E+03	8.8040E+02	1.1391E+03	1.1707E+03
RC 2 Econ Cost ($/yr)	7.3166E+02	9.6137E+02	9.2244E+02	8.3018E+00	9.3496E+02	9.3496E+02
RC 3 Econ Cost ($/yr)	2.5268E+03	3.2302E+03	2.8889E+03	2.6634E+03	3.4517E+03	3.5414E+03
Total Econ Cost ($/yr)	4.1656E+03	5.3031E+03	4.8714E+03	3.5521E+03	5.5258E+03	5.6470E+03
Off-Site Exposure Cost						
Zpha	$4,803	$ 6,115	$ 5,587	$ 4,345	$6,409	$6,557
Wpha	$ 64,446	$ 82,046	$74,957	$ 58,295	$ 85,984	$ 87,969
Off-Site Economic Cost						
Zea	$ 4,166	$ 5,303	$ 4,871	$ 3,552	$ 5,526	$5,647
Wpha	$ 55,890	$71,152	$65,360	$47,658	$ 74,139	$ 75,766
On-Site Exposure Cost						
WIO	$700	$1,340	$1,322	$ 1,013	$ 1,273	$ 1,357
WLTO	$3,050	$ 5,839	$5,763	$ 4,414	$ 5,549	$ 5,914
Total	$ 3,749	$ 7,178	$ 7,085	$5,427	$6,822	$ 7,271
On-Site Economic Costs						
Cleanup and Decontamination						
PVCD	$1,078,745,778	$ 1,078,745,778	$ 1,078,745,778	$ 1,078,745,778	$1,078,745,778	$ 1,078,745,778
UCD ($-yr)	$14,473,531,133	$14,473,531,133	$14,473,531,133	$14,473,531,133	$ 14,473,531,133	$ 14,473,531,133
Replacement Power Cost						
PVRP	$ 1,927,554,493	$1,927,554,493	$1,927,554,493	$1,927,554,493	$1,927,554,493	$ 1,927,554,493
URP ($-yr)	$24,289,327,255	$ 24,289,327,255	$24,289,327,255	$24,289,327,255	$ 24,289,327,255	$ 24,289,327,255
Total On-Site Economic Costs ($)	$ 306,294	$586,407	$ 578,805	$ 443,355	$557,272	$593,937
Total Cost of Severe Accident Risk	$430,379	$ 746,782	$726,207	$ 554,736	$ 724,217	$ 764,943
External Event Multiplier (2)	860,758.96	1,493,564.91	1,452,413.13	1,109,471.80	1,448,433.06	1,529,885.59
Change from Base	$(675,053)	$ (42,247)	$ (83,399)	$(426,340)	$ (87,379)	$ (5,926)
Costs (from TVA)	$ 31,675	$ 9,126,460	$31,675	$ 19,608	$439,945	$ 615,605
CB Ratio	-21.31	0.00	-2.63	-21.74	-0.20	-0.01

c. *NRC Request*

As indicated in the operating license SER, TVA will align the licensing and design bases on the WBN Units 1 and 2 to the fullest extent practicable. Thus, any changes made as a result of the Unit 2 SAMA analysis would be expected to also be made in Unit 1. As a result, the "per unit" cost of implementing certain SAMAs would be less than if implemented on only a single unit. This could significantly impact the cost for certain SAMAs (e.g., procedure changes). Identify the SAMAs that might be implemented on both units and discuss the impact of sharing the implementation costs on the results of the cost benefit analysis for Unit 2.

TVA Response

Of the Phase II SAMAs, only SAMA 8 (increase training on response to loss of two 120V AC buses which causes inadvertent actuation signals) has a benefit-cost ratio within a factor of 2 of being cost effective.

It should be noted that incorporation of this SAMA at both units will require modification of the Unit 1 training .While the cost will be reduced per unit, the total will still be more than the current $26,773 estimated cost.

d. *NRC Request*

The benefit for SAMA 8, "Increase training on response to loss of two 120V AC buses which causes inadvertent actuation signals," was estimated by eliminating the consequences of each of the single bus initiating events. Presumably this was done because the model did not include a two bus failure initiating event. Confirm that the loss of two bus initiating event would have similar or less impact than the loss of the single buses individually.

TVA Response

The loss of more than one bus is judged to be adequately addressed by the single bus initiating event, which includes the consequential loss of additional buses.

e/f. *NRC Request*

The description of SAMA 46, "Add a service water pump," indicates that an alternate pump exists that could be temporarily connected to the ERCW, and that a permanent diesel-driven 10,000 gpm pump could be installed at the intake pumping station flush connection to the ERCW. It is not clear which of these two options was evaluated. Clarify and provide an assessment of both options.

TVA Response

As a part of the enhancements to security features, an alternate temporary pump exists that can be connected to the ERCW system. The Phase II cost benefit study determined that the permanent installation of the pump is not cost beneficial.

g. *NRC Request*

The enhancement evaluated for SAMA 156, "Eliminate RCP thermal barrier dependence on CCW such that loss of CCW does not result directly in core damage," was a procedure change that was found to be cost-beneficial based on a bounding assumption that RCP seal injection is always successful when AC power is available. However, a procedure change would not realistically provide this level of risk reduction. Discuss whether there is another enhancement (hardware modification) that might be more effective in reducing the risk than a procedure change and still cost effective.

TVA Response

See SAMA 58 – install improved reactor coolant pump seals. WBN already has high temperature RCP seals installed at Unit 2, and it is unclear as to the cost and time frame for availability of a current concept seal package that would prevent LOCA on loss of cooling and injection. See response to RAI 5.h.x. Also, if AC power is not available, the RCPs would lose motive power, such that the likelihood of seal failure is much reduced from pump operation with no seal cooling.

h. *NRC Request*

For a number of the Phase II SAMAs evaluated in Section 8, the information provided does not sufficiently describe the associated modifications and what is included in the cost estimate. Provide a more detailed description of both the modification and the cost estimate for Phase II SAMAs 32, 56, 87, 273, and 280.

TVA Response

The cost estimates include the cost for engineering, construction and materials.

SAMA 32 (add the ability to automatically align ECCS recirculation to recirculation mode on RWST depletion). This includes design and field installation of an automatic transfer of the safety injection suction to the associated RHR pump discharge ("piggyback" mode). Cost Estimate $1.5M

SAMA 56 (install independent RCP seal injection system without dedicated diesel) includes design and installation of a separate, high pressure, low volume pump for RCP seal injection in the room that was previously used for a positive displacement pump. Costs include removal of the positive displacement pump and installation of new pump. Cost Estimate $4M

SAMA 87 (replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans to eliminate instrument air system dependence on service water) evaluates replacement of the current three 50% capable air compressors with air cooled units.
Cost Estimate $886K

SAMA 273 (provide redundant path for ECCS suction from the RWST around check valve 62-504) includes design and installation of separate parallel piping with separate check valve to require a second failure to prevent flow, making this pinch point more reliable. Cost Estimate $440K

SAMA 280 (add new Unit 2 air compressor similar to the Unit 1 D compressor to improve air system availability) evaluates the addition of a new, full capacity (100%) air compressor to the current installation. The new unit would be powered from existing non-essential power. Cost Estimate $387K

7. <u>*NRC Request*</u>

Provide the following information with regard to the sensitivity and uncertainty analyses:

a. *Provide an assessment of whether any of the Phase I SAMAs screened due to excessive implementation costs or very low benefit should be retained for a Phase II evaluation based on the 95th percentile results for CDF and LERF. Provide a Phase II evaluation for any retained SAMAs.*

<u>TVA Response</u>

This is similar to the uncertainty analysis performed in Section 9.2 of the report for Phase II SAMAs. As noted in that section, the 95% CDF increases the point estimate mean value by a factor of 2.78 (178% increase) and LERF by a factor of 2.5 (150% increase). This compares with the increase of 135% noted in the response to RAI 3.c (above) for increase in external event CDF and LERF. Therefore, the breakpoint cost value for this uncertainty evaluation would be $4,269,557.

For reference, the SAMAs that were not initially screened from further consideration on cost in RAI 3.c are listed below. For those that are not listed in Tables 11, 12, 13, or 14 (RRW 1.02 or greater), the maximum abated risk can be approximated as no higher than 2% x 2.78 x $1,535,812 bounding cost = $85,391. This effectively removes any significant hardware or safety analysis changes from consideration.

1. SAMA 2, replace lead-acid batteries with fuel cells, compares with the $300,000 cost to only replace batteries I and II for Unit 1 (see discussion under SAMA 174 in Table 16). From Table 11, DC power has a RRW of 1.047 for CDF, such that the potential abated risk at 95% CDF is 4.7% x 2.78 x $1,535,812 bounding cost = $200,669. This gives a benefit to cost ratio of ($200,669 / $300,000 =) 0.67, such that the proposed SAMA is not cost

effective. This estimate is conservative because it does not include the cost of supplying the fuel cell for the 40 year life of the plant.

2. SAMA 13, add an additional buried offsite power source. From the response to RAI 1.a above, LOSP has a contribution of 3.0%, such that the maximum risk abatement would be 0.030 x 2.78 x $1,535,812 bounding cost = $128,086. This makes the hardware installation cost for burial of one of the 161kV offsite lines prohibitive relative to the potential risk reduction.
3. SAMA 26, provide additional high pressure injection pump with independent diesel. From Table 11, safety injection has a RRW of less than 1.02 for CDF. This gives a maximum risk abatement of 0.02 x 2.78 x $1,535,812 bounding cost = $85,391. This makes the design, safety analysis, change to the licensing basis, and installation of the pump and generator prohibitive relative to the potential risk reduction.
4. SAMA 28, add diverse low pressure injection system to improve injection capability. RHR pump top events RA and RB have a maximum RRW of 1.0653 for CDF, such that the maximum risk abatement for this proposed SAMA is 0.0653 x 2.78 x $1,535,812 bounding cost = $278,802. This makes the design, safety analysis, change to the licensing basis and installation of the change prohibitive relative to the potential risk reduction.
5. SAMA 34, provides in-containment RWST, eliminating need for swap over to sump recirculation. As noted in the Phase I Comment, there is limited room inside the containment for a tank of this size. High pressure recirculation top event RRH has an RRW of 1.3809 for CDF. This gives a maximum risk abatement of 0.381 x 2.78 x $1,535,812 bounding cost = $1,626,701. This makes the design, safety analysis, change to the licensing basis, and installation of the new RWST inside containment marginal relative to the potential risk reduction, provided that the tank volume can physically be located inside containment.
6. SAMA 55, install independent RCP seal injection system with dedicated diesel (CVCS supply top event VS has RRW for CDF of 1.0731, such that the maximum benefit would be 0.073 x 2.78 x $1,535,812 bounding cost = $311,678. This makes the design, safety analysis, change to the licensing basis, and installation of the new system and dedicated diesel generator prohibitive relative to the potential risk reduction.
7. SAMA 95, enhances fire protection system and standby gas treatment system hardware and procedures to improve fission product scrubbing in severe accidents. As noted in Phase 1 Comments, EPSIL contains instructions for spraying release points with fire water. This would provide fission product scrubbing, such that the SAMA would provide minimal additional benefit.
8. SAMA 99, strengthen primary/secondary containment (add ribbing to containment shell). From Table 12, containment isolation has a RRW for LERF of 1.062. This gives a maximum risk abatement of 0.062 x 2.50 x $1,535,812 bounding cost = $238,051. This makes the design, safety analysis, change to the licensing basis, and installation of the change prohibitive relative to the potential risk reduction.
9. SAMA 105, delay containment spray actuation during LOCA. From Tables 11 and 12, containment spray contributes less than 2.0% to CDF and to LERF. Therefore, the maximum risk abatement would be 0.02 x 2.50 x

$1,535,812 bounding cost = $76,790. This makes the safety analysis and change to the licensing basis prohibitive relative to the potential risk reduction.

10. SAMA 106, install automatic containment spray pump throttle valves containment spray. From Tables 11 and 12, containment spray contributes less than 2.0% to CDF and to LERF. Therefore, the maximum risk abatement would be 0.02 x 2.50 x $1,535,812 bounding cost = $76,790. This makes the safety analysis and installation cost prohibitive relative to the potential risk reduction.
11. SAMA 109, installs passive hydrogen control system. From Table 14, the current active hydrogen control system has a RRW of less than 1.02 for LERF. This gives a maximum risk abatement of 0.02 x 2.50 x $1,535,812 bounding cost = $76,790. This makes the system design, safety analysis, and installation cost prohibitive relative to the potential risk reduction.
12. SAMA 110, erects a barrier to prevent containment failure due to core melt ejection. From Table 12, containment isolation systems have a RRW of 1.062, such that the maximum risk abatement would be 0.062 x 2.50 x $1,535,812 bounding cost = $238, 051. This makes the hardware design, safety analysis update, and installation cost prohibitive relative to the potential risk reduction.
13. SAMA 122, installs redundant spray system to depressurize the primary system during SGTR. From the response to RAI 1.a above, SGTR has a contribution of 0.42%, such that the maximum risk abatement would be 0.0042 x 2.78 x $1,535,812 bounding cost = $17,932. This makes the hardware installation cost prohibitive relative to the potential risk reduction.
14. SAMA 124, provides improved instrumentation to detect SGTR, such as Nitrogen 16 monitors. From the response to RAI 1.a above, SGTR has a contribution of 0.42%, such that the maximum risk abatement would be 0.0042 x 2.78 x $1,535,812 bounding cost = $17,932. This makes the hardware installation cost prohibitive relative to the potential risk reduction.
15. SAMA 129, vent main steam safety valves into containment to reduce consequences of SGTR. From the response to RAI 1.a above, SGTR has a contribution of 0.42%, such that the maximum risk abatement would be 0.0042 x 2.78 x $1,535,812 bounding cost = $17,932. This makes the hardware installation cost prohibitive relative to the potential risk reduction.
16. SAMA 147, installs digital large break LOCA protection system. From the response to RAI 1.a above, total contribution to CDF from excessive LOCA (ELOCA) =1.7% and from large LOCA (LLOCA) = 0.19%, or 1.9% total. This gives a maximum risk abatement of 0.019 x 2.78 x $1,535,812 bounding cost = $81,121. This makes the hardware installation cost prohibitive relative to the potential risk reduction.
17. SAMA 175, create lake water backup for EDG cooling. From the response to RAI 1.a above, LOSP has a contribution of 3.0%, such that the maximum risk abatement would be 0.030 x 2.78 x $1,535,812 bounding cost = $128,086. This makes the design, safety analysis update, and hardware installation cost for the new diesel generator cooling line to the dam impound prohibitive relative to the potential risk reduction.
18. SAMA 191, provides self-cooled ECCS seals. As stated in the Phase 1 Comment, charging and safety injection pumps at WBN have mechanical

seals. Only the RHR pump seals are cooled by CCS. Also, CCS cools the RHR heat exchangers and the lube oil coolers for charging and safety injection, such that ECCS would still be failed due to dependency following a loss of CCS. Therefore, this SAMA would provide minimal incremental risk reduction, even considering 95% CDF and LERF.

19. SAMA 215, provides RCP seal cooling to prevent seal LOCA during SBO. From the response to RAI 1.a, SBO contributes 2.2% of CDF, such that maximum risk abatement would be 0.022 x 2.78 x $1,535,812 bounding cost = $93,930. Given that the change would include a black start power supply for the seal cooling pump, this makes the hardware design and installation cost prohibitive relative to the potential risk reduction.
20. SAMA 242, permanent, dedicated generator for the NCP with local operation of TD AFW after battery depletion. This is bounded by TD AFW pump failure to start, which has a RRW for LERF of 1.0307 from Table 14. This gives a maximum risk abatement of 0.0307 x 2.50 x $1,535,812 bounding cost = $117,873. This makes the design and hardware installation cost for the new generator prohibitive relative to the potential risk reduction.
21. SAMA 255, permanent, dedicated diesel generator for the NCP, one motor driven AFW and a battery charger. This is bounded by motor driven AFW pump failure to start, which has a RRW for CDF and LERF of less than 1.02 from Tables 13 and 14. This gives a maximum risk abatement of 0.02 x 2.78 x $1,535,812 bounding cost = $85,391. This makes the design and hardware installation cost for the new generator prohibitive relative to the potential risk reduction.
22. SAMA 270, delay containment spray actuation relative to Phase B isolation. Since containment spray contributes less than 2.0% of LERF (from Table 12) the maximum risk abatement would be 0.02 x 2.50 x $1,535,812 bounding cost = $76,790. This makes the safety analysis and installation cost prohibitive relative to the potential risk reduction.
23. SAMA 274, replaces CCS pumps with positive displacement pumps. Table 11 gives a RRW of 1.067 for CCS, such that the maximum risk abatement would be 0.067 x 2.78 x $1,535,812 bounding cost = $286,060. This makes the hardware installation cost prohibitive relative to the potential risk reduction.
24. SAMA 282, provides cross-tie to Unit 1 RWST to extend RWST capacity. RWST makeup top event OMU has an RRW of 1.0159, such that the maximum risk abatement for this SAMA at the 95% CDF value would be 0.0159 x 2.78 x $1,535,812 bounding cost = $67,886. This makes the analysis, design change, and hardware installation cost prohibitive relative to the potential risk reduction.

The 22 SAMAs that were screened from further consideration in Phase I on low benefit were:

1. SAMA 5, provide DC bus crossties. Even considering the 95% multiplier, this was judged to remain a marginal improvement considering the system crosstie capability on the 480-VAC supply side and availability of a spare #5 battery.

2. SAMA 29, provides capability for alternate injection via diesel-driven fire pump. This is considered for site flooding scenarios, but considered to be a marginal improvement due to extreme cooldown necessary to take plant from normal operation to 100 to 150 psig to enable fire protection injection.
3. SAMA 47, enhances screen wash system. Screen failure contributes less than 2% of CDF and would remain a minimal contributor even when considering 95% CDF and LERF.
4. SAMA 50, enhances loss of CCW procedure to underscore RCS cooldown prior to seal LOCA. As noted in Phase 1 Comment, AOI-15 and AOI-24 already require RCS cooldown after isolation of the CCS path to the RCP thermal barrier and isolation of RCP seal injection, such that additional procedural changes would have minimal risk improvement, even considering 95% CDF and LERF
5. SAMA 53, on loss of ERCW, proceduralizes shedding CCW loads to extend CCW heat up time. As stated in the Phase 1 Comment, while AOI-13 does not direct the operator to the loss of CCS procedure, it does include tripping the RCPs, isolation of thermal barrier cooling, cooling down the plant and cross tying ERCW if available, such that this SAMA is effectively already implemented in the plant.
6. SAMA 79, replace existing PORVs with larger units, so only one is necessary for bleed and feed cooling. As stated in the Phase 1 Comments, a single PORV is adequate for bleed and feed cooling when charging is available, such that a second PORV (and this SAMA) would only be required when safety injection and NOT charging is available. This set of circumstances is judged to remain a minimal contributor to CDF even considering the 95% multiplier.
7. SAMA 80, provide redundant train or means of ventilation, is judged to provide minimal risk reduction given the considerations listed in the Phase 1 Comment for this SAMA, even considering 95% CDF and LERF.
8. SAMA 81, add diesel building high temperature alarm or redundant louver and thermostat. As noted in the Phase 1 Comments, an operator is stationed in the DG building during a DG start, such that the proposed SAMA would provide minimal improvement in detection capability for failure of DG room ventilation even considering 95% CDF and LERF.
9. SAMA 92, use fire water system as a backup source for containment spray. As noted in the Phase 1 Comment, this SAMA would not provide containment heat removal (only add additional inventory to the containment sump) and would provide minimal flow and fission product heat removal, such that impact would be minimal, even considering 95% CDF and LERF.
10. SAMA 116, ensures ISLOCA releases are scrubbed, would remain a minimal impact due to the considerations noted under Phase 1 Comments even considering 95% CDF and LERF.
11. SAMA 137, provides capability to remove power from bus powering control rods. See SAMA 136 – install motor generator set trip breakers in control room, which was retained for Phase II evaluation.
12. SAMA 152, develop procedures for transportation and nearby facility accidents, would remain a minimal impact for the reasons stated under Phase 1 Comments (anti-barge boom installed at plant and no identified

hazardous barge shipments near the site) even considering 95% CDF and LERF.

13. SAMA 167, enhances air return fans. As stated in Phase I Comments, CAR fans provide only a negligible contribution to ability to contain hydrogen burn such that this remains a minimal contributor even when considering 95% CDF/LERF.
14. SAMA 183, implements internal flood prevention and mitigation enhancements, including use of submersible MOV operators. As shown in the response to RAI 1.a, above, only intake floods FLPH1A (1.6%) and FLPH1B (2.8%) contribute more than 0.04% of CDF. The primary impact of these initiating events is related to a loss of the associated train of ERCW, which would be unaffected by the proposed change, such that the proposed SAMA would have no material impact on risk reduction at WBN.
15. SAMA 184, implements internal flood enhancements identified at Fort Calhoun. As noted in the discussion of this SAMA, the contributors identified at Fort Calhoun are minimal or nonexistent (AFW flood leading to need to remove watertight door) at WBN and would remain so even considering 95% CDF/LERF.
16. SAMA 199, provides Auxiliary Building vent/seal structure to enhance ventilation. AB ventilation is not a material contributor to CDF at WBN and would remain minimal even considering 95% CDF and LERF.
17. SAMA 222, establish preventive maintenance program for expansion joints, bellows and boots. Limited use of boots at WBN leads to minimal contribution to flooding from these types of failures, even considering 95% CDF/LERF multiplier.
18. SAMA 225, upgrade main turbine controls. Turbine trip contributes less than 1% of CDF, and would remain minimal contributor using 95% multiplier.
19. SAMA 234, implements automatic initiation of HPI on low RCS level. Due to pressure drop on loss of RCS inventory, HPI would actuate on low pressure, such that this additional start would provide minimal improvement in risk (only if low pressure signal did not actuate) even considering 95% multiplier.
20. SAMA 254, alternate fuel oil tank with gravity feed capability. Fuel oil contributes 2% of CDF, such that this would remain a minimal contributor even considering 95% multiplier.
21. SAMA 262, provides connections for centrifugal charging pumps to ERCW. Judged to provide minimal improvement in risk given that plant design currently has this feature for the A pump.
22. SAMA 277, replaces shutdown board chillers. Chillers provide less than 2% of CDF and are judged to remain minimal contributor to CDF even with 95% multiplier.

b. <u>*NRC Request*</u>

Provide the dollar benefit value for each of the Phase II SAMAs using the 3 percent discount rate.

TVA Response

	Table 17 Phase II Analysis Results with 3% Discount Rate		
SAMA Number	*SAMA Title*	*Estimated Benefit with 7% Discount Rate*	*Estimated Benefit with 3% Discount Rate*
4	Improve DC bus load shedding.	$83,399	$148,873
8	Increase training on response to loss of two 120V AC buses which causes inadvertent actuation signals.	$21,469	$37,482
SAMA Number	*SAMA Title*	*Estimated Benefit with 7% Discount Rate*	*Estimated Benefit with 3% Discount Rate*
32	Add the ability to automatically align emergency core cooling system to recirculation mode upon refueling water storage tank depletion.	$530,264	$902,089
45	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	$89,003	$161,543
46	Add a service water pump.	$102,000	$208,502
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel.	$675,053	$1,224,044
70	Install accumulators for turbine-driven auxiliary feedwater pump flow control valves.	$1,945	$3,520
71	Install a new condensate storage tank (auxiliary feedwater storage tank).	$0	$0
87	Replace service and instrument air compressors with more reliable compressors which have self-contained air cooling by shaft driven fans.	$121,460	$177,241
112	Add redundant and diverse limit switches to each containment isolation valve.	$4,565	$8,263
136	Install motor generator set trip breakers in control room.	$7,397	$24,180
156	Eliminate RCP thermal barrier dependence on CCW, such that loss of CCW does not result directly in core damage.	$675,053	$1,235,325
176	Provide a connection to alternate offsite power source.	$42,247	$76,467
243	Modify the Controls and Operating Procedures for Hydro Station to allow for Rapid Response.	$83,399	$148,873
256	Install Fire Barriers Around Cables or Reroute the Cables Away from Fire Sources.	$426,340	$764,712
273	Provide a redundant path for ECCS suction from the RWST around check valve 62-504.	$87,379	$175,978
276	Provide an auto start signal for AFW on loss of Standby Feedwater pump.	$5,926	$10,726
279	Provide a permanent tie-in to the construction air compressor.	$121,460	$215,688
280	Add new Unit 2 air compressor similar to the Unit 1 D compressor.	$121,460	$231,941

8. *NRC Request*

For certain SAMAs considered in the SAMA submittal, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, provide an evaluation of the following SAMAs:

a. *Purchasing or manufacturing a "gagging device" that could be used to close a stuck-open SG safety valve for a SGTR event prior to core damage.*

TVA Response

The use of a gagging device to close a stuck-open SG safety valve is not practical at WBN due to the design and location of these valves in the plant. Implementing such a device would involve a significant hazard to plant personnel.

b. *NRC Request*

Utilizing the spare fifth diesel generator mentioned in the disposition of SAMA 261 without going through the expense of complete refurbishing and licensing.

TVA Response

While restoration, refurbishment, and licensing a class 1E generator contributes significantly to the estimated $5 million cost of making the DG operable (see response to SAMA 9), the unit has been cannibalized to the point that any new unit would be entirely new. Also, the new installation would have to include class 1E interface to the shutdown boards in the design and failure modes analysis.

c. *NRC Request*

Providing procedures and cabling to enable the use of the trailer-mounted 2 MW diesel generator provided in response to GSI-189 to be used to power selected equipment such as battery chargers, and/or individual pumps.

TVA Response

WBN has a 2 Mw diesel generator which is used for GSI-189 that can be used according to plant procedures to power certain pumps in the plant.

d. *NRC Request*

Purchasing and installing a permanent diesel-generator to supply power to the normal charging pump.

TVA Response

Installation of diesel generator to supply power to the normal charging pump would have to consider power supply selection arrangement and interface with normal supply from shutdown board power. Physical space limitations in the plant would prohibit placing this diesel generator near the pump. Significant cable routing would have to be accomplished to implement this change in addition to the procedure changes and training involved.

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Watts Bar Nuclear Plant Severe Reactor Accident Analysis

Prepared for:

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May 30, 2007

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The opinions and recommendations contained in this Report are dependant on the accuracy, completeness and correctness of the data, specifications, documents and other information provided by Customer, whether provided in writing or orally ("Customer Information"). Customer Information may include information, data and documents relating to Customer's [facility, personnel, systems, equipment, software, electronic data, and procedures and policies]. If any of the Customer Information is inaccurate, incomplete or incorrect, the opinions and recommendations of SAIC contained in the Report cannot be relied upon by Customer. Also, SAIC cannot guarantee or assure that risks and threats other than those addressed in this report will not occur.

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Watts Bar Nuclear Plant Severe Reactor Accident Analysis

Executive Summary

Tennessee Valley Authority (TVA) is preparing a Supplemental Environmental Impact Statement for the Watts Bar Nuclear (WBN) Plant site that includes future operation of Watts Bar Unit 2. This analysis was performed to estimate the human health impacts from potential accidents at the site. The term "accident" refers to any unintentional event (i.e., outside the normal or expected plant operation envelope) that results in a release or a potential for a release of radioactive material to the environment. The Nuclear Regulatory Commission (NRC) categorizes accidents as either design basis or severe. Design basis accidents are those for which the risk is great enough that NRC requires plant design and construction to prevent unacceptable accident consequences. Severe accidents are those that NRC considers too unlikely to warrant design controls.

TVA maintains a probabilistic safety assessment model to use in evaluating the most significant risks of radiological release from WBN fuel into the reactor and from the reactor into the containment structure. In 1995, both TVA and NRC concluded that, except for a few procedural changes implemented as part of the WBN operation, none of the Severe Accident Mitigation design alternatives were beneficial to further mitigating the risk of severe accidents. Since then, TVA has implemented the industry-required design and corresponding mitigating action changes as required by NRC for continued operation of WBN Unit 1. and is expected to implement them for operation of Unit 2. The design changes have already been implemented in the WBN Unit 1 probabilistic safety assessment model. The analysis is based on the WBN Unit 1 probabilistic safety assessment model, which considered applicable for Unit 2 operations because of its similarity to Unit 1. Only severe reactor accident scenarios leading to core damage and containment bypass or failure are considered, here. Accident scenarios that do not lead to containment bypass or failure are not presented because the public and environmental consequences would be significantly less.

The MACCS2 computer code (Version 1.13.1) was used to perform probabilistic analyses of radiological impacts. The generic input parameters given with the MACCS2 computer code that were used in the NRC's severe accident analysis (NUREG-1150) formed the basis for the analysis. These generic data values were supplemented with parameters specific to Watts Bar nuclear plant and the surrounding area. Site-specific data included population distribution, economic parameters, and agricultural product. Plant-specific release data included nuclide release, release 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 emergency planning zone (EPZ) evacuation time. These data in combination with site specific meteorology were used to simulate the probability distribution of impact risks (exposure and fatalities) to the surrounding 80-kilometer (within 50 miles) population.

Table ES-1 summarizes the consequences of the beyond design-basis accident, with mean meteorological conditions, to the maximally exposed offsite individual, and an average individual member of population residing within an 80-kilometer (50-mile) radius of the reactor site. The analysis assumed that a site emergency would have been declared early in the 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 have been implemented to initiate evacuation of 99.5 percent of the public within 16 kilometers (10 miles) of the plant. The location of the maximally exposed offsite individual may or may not be at the site boundary for these accident sequences because emergency action guidelines would have been

implemented and the population would be evacuating from the path of the radiological plume released by the accident.

Table ES–1 Severe Reactor Accident Annual Risks

Release Category (frequency per reactor year)	Maximally Exposed Offsite Individual		Average Individual Member of Population within 80 Kilometers (50 miles)	
	Dose Risk [a] (rem/year)	Cancer Fatality [b]	Dose Risk [a] (rem/year)	Cancer Fatality [b]
I - Early Containment failure (3.4×10^{-7})	2.2×10^{-5}	2.6×10^{-8}	1.8×10^{-7}	1.1×10^{-10}
II - Containment Bypass (1.4×10^{-6})	2.2×10^{-5}	1.3×10^{-8}	8.2×10^{-7}	4.9×10^{-10}
III - Late Containment Failure (3.0×10^{-6})	4.6×10^{-7}	2.8×10^{-10}	1.3×10^{-7}	7.8×10^{-11}

[a] Includes the likelihood of occurrence of each release category.
[b] Increased likelihood of cancer fatality per year.

The results presented in this table indicates that the highest risk to the maximally exposed offsite individual is one fatality every 38 million years (or 2.6×10^{-8} per year) and the highest risk to an average individual member of the public is one fatality every 2 billion years (or 4.9×10^{-10} per year). Overall, the risk results presented above are small. Completion and operation of WBN Unit 2 would not change the risks evaluated here because the likelihood of an accident that could affect both units and lead to radioactive releases beyond those analyzed here would be extremely low. This is consistent with the conclusions of NRC's Generic Environmental Impact Statement for License Renewal of Nuclear Plants, (GEIS). Accidents that could affect multiunit sites are initiated by external events. Severe accidents initiated by external events as tornadoes, floods, earthquakes, and fires traditionally have not been discussed in quantitative terms in final environmental statements and were not considered in the GEIS. 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.

WATTS BAR NUCLEAR PLANT SEVERE REACTOR ACCIDENT ANALYSIS

1. Introduction

Tennessee Valley Authority (TVA) is preparing a supplemental environmental impact statement for the Watts Bar Nuclear (WBN) Plant site that includes future operation of Watts Bar Unit 2. This analysis is being performed to estimate the human health impacts from potential accidents at the site. The term "accident" refers to any unintentional event (i.e., outside the normal or expected plant operation envelope) that results in a release or a potential for a release of radioactive material to the environment. The Nuclear Regulatory Commission (NRC) categorizes accidents as either design-basis or severe. Design-basis accidents are those for which the risk is great enough that NRC requires plant design and construction to prevent unacceptable accident consequences. Severe accidents are those that NRC considers too unlikely to warrant design controls.

TVA maintains a probabilistic safety assessment (PSA) model to use in evaluating the most significant risks of radiological release from WBN fuel into the reactor and from the reactor into the containment structure. For the WBN Unit 1 Severe Accident Mitigation Design Alternative (SAMDA) analysis conducted in 1995, TVA used the PSA model output as input to an NRC-approved model that calculated economic costs and dose to the public from hypothesized releases from the containment structure into the environment. Using regulatory analysis techniques, TVA calculated the monetary value of the unmitigated WBN severe accident risks. TVA and NRC concluded that, except for a few procedural changes implemented as part of the WBN operation, none of the SAMDAs were beneficial to further mitigating the risk of severe accidents (NRC 1995). Since then, TVA has implemented the industry-required equipment design and corresponding mitigating action changes required by NRC for continued operation of WBN Unit 1 and is expected to implement them for operation of Unit 2. Therefore, prior to operation of Unit 2, the plant will meet all required designs and conditions for mitigating the risk of severe accidents.

Based on the statement of the work (TVA 2007), the analysis herein will follow a method similar to that used in the *Final Environmental Impact statement for the Production of Tritium in a Commercial Light Water Reactor* (CLWR EIS) (DOE 1999). TVA's analyses of design-basis accidents are described in the WBN Updated Final Safety Analysis Report and are not within the scope of this analysis. This analysis is limited to severe reactor accidents. The analyses presented here are based on the WBN Unit 1 PSA model, which is considered applicable for Unit 2 operations, because of the similarity to Unit 1 operations. Only severe reactor accident scenarios leading to core damage and containment bypass or failure are considered here. Accident scenarios that do not lead to containment bypass or failure are not presented because the public and environmental consequences would be significantly less. Three modes of containment failures are defined: containment bypass, early containment failure, and late containment failure (see **Table 1**).

The magnitude of the radioactive release to the atmosphere resulting from an accident depends on the timing of the reactor vessel failure and the containment failure. Source terms associated with various release categories describe the fractional releases for representative radionuclide groups, as well as the timing, duration, and energy of potential releases.

Table 1 Definition and Causes of Containment Failure Mode Classes

Failure mode	Definition and Causes
Containment 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.
Early Containment Failure	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.
Late Containment Failure	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, hydrogen combustion, and basemat melt-through by core debris. Venting containment late in the accident also is classified as a late containment failure.

2. Representative Severe Reactor Accident Scenarios

Plant damage states 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 representative accident scenarios are limited to the dominant sequence or sequences within a plant damage state that are major contributors to the release level categories associated with each of the containment failure modes defined above. The information is based on TVA's most recent analysis of severe accidents performed under the individual plant examination program, which covers both the level 1 and level 2 probabilistic risk assessments in detail. TVA's analyses of the Watts Bar and Sequoyah individual plant examinations were submitted to NRC in September 1992 (TVA 1992a, TVA 1992b). Both of these analyses have been revised since (TVA 1995, TVA 1994).

The selected release categories and examples of various accident scenarios leading to containment failure and/or bypass are presented below. Release Category I results from a reactor vessel breach with 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. **Table 2** shows the equilibrium reactor core nuclide inventory at the time of a reactor trip (TVA 2007). **Table 3** provides important information on time to core damage, containment failure, release duration, and the isotope release fractions associated with each of the release categories (TVA 2007). **Table 4** provides a representation of the dominant accident scenarios that lead to each release category and the likelihood of their occurrence (TVA 2007).

Table 2 Watts Bar Unit 1 Core Inventory

Nuclide	*Isotope*	*Group* [a]	*Curies* [b]
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
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

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

[a] The grouping is based on NUREG-1465.
[b] Source: TVA 2007.

Table 3 Release Category Timing and Source Terms

Release Times, Heights, Energies, and Source Terms for Selected Release Categories					
Release Category	***Release Height (meters)***	***Warning Time (hours)***	***Release Time (hours)***	***Release Duration (hours)***	***Release Energy [a] (megawatts)***
I	10.00	8	10	2	28
II	10.00	20	24	4	1
III	10.00	20	30	10	3.5

Fission Product Source Terms (fraction of total inventory)										
Release Category	***NG***	***I***	***Cs***	***Te***	***Sr***	***Ru***	***La***	***Ce***	***Ba***	***Mo***
I	0.90	0.042	0.043	0.044	0.0027	0.0065	0.00048	0.004	0.0046	0.0065
II	0.91	0.21	0.19	0.0004	0.0023	0.07	0.00028	0.00055	0.025	0.07
III	0.94	0.0071	0.011	0.0052	0.00036	0.00051	4.2×10^{-6}	4.0×10^{-6}	0.0013	0.00051

NG = Noble gases.
[a] These values were taken from similar accident scenarios given in NUREG/CR-4551.
Sources: TVA 1992a, TVA 1992b, TVA 2007.

Table 4 Release Category Frequencies and Related Accident Sequences for the Watts Bar Nuclear Plant

Release Category	*Frequency*	*Remarks (Example Scenario)*
I	3.4×10^{-7}	The major accident contributors to this release event are initiated by loss of offsite power and the essential raw cooling water system: failure of the emergency diesels to start and/or failures in the 125-volt direct current distribution system, together with loss of secondary cooling; and no recovery before core melt.
II	1.4×10^{-6}	The main contributor to this release event is initiated by a steam generator tube rupture in conjunction with either an operator error or a random failure of electrical distribution systems, leading to failure of the coolant system and failure to control the affected steam generator before core melt occurs.
III	3.0×10^{-6}	The major accident contributors to this release event are initiated by loss of offsite power and various failures in the alternating current distribution systems; no recovery of power before core melts; a reactor coolant system loss-of-coolant accident (large- and medium-sized loss-of-coolant accident); and failure to establish long-term core cooling.

Source: TVA 2007.

3. Methodology for Estimating Radiological Impacts

3.1 Introduction

The MACCS2 computer code (Version 1.13.1) was used to perform probabilistic analyses of radiological impacts. 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).

The input parameters given with the MACCS2 Sample Problem A, which include the data used in NUREG 1150 (NRC 1998), formed the basis for the analysis. These generic data values were supplemented with parameters specific to the WBN Plant and the surrounding area. Site-specific data included population distribution, economic parameters, and agricultural product. Plant-specific release data included nuclide release, release 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 emergency planning zone (EPZ) evacuation time. These data, in combination with site-specific meteorology, were used to simulate the probability distribution of impact risks (exposure and fatalities) to the surrounding 80-kilometer (within 50 miles) population.

3.2 Site-Specific Parameters

This section describes the method and assumptions used to develop site-specific parameters.

Population

The population surrounding the WBN Plant site was estimated for the year 2040. The distribution is given in terms of the population at 10 distances, ranging from 0 miles to 50 miles from the plant, 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 (Eblen 2007). The total projected population within 50 miles of the site was estimated to be 1,523,390, (see **Table 5**).

Agriculture and Economy

Agriculture production information was generated using SECPOP 2000. SECPOP provides the MACCS2 model with required information on the crops season and shares (fraction of land devoted to the crop). The SECPOP-generated data were compared with those used in the CLWR EIS, which was based on data for neighboring counties. The SECPOP data were considered more representative (more recent) and, except for the pasture, use larger land fractions for specific crops.

MACCS2 also requires spatial distribution of certain economic data (fraction of land devoted to farming, annual farm sales, fraction of farm sales resulting from diary production, property values of farm and nonfarm land). SECPOP also produces this data for each site. Although these data were generated and added to the site data, they were not used in the analysis.

Table 5 Projected 2040 Population Distribution within 80 Kilometers (50 miles) of Watts Bar Nuclear Plant

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	25,3831	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	106.392	201.482
SW	0	0	0	0	0	1.051	12.978	14,499	62,307	111.795	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	2.472	4.930	19.286	11.3370	299.816	353.432	728.967	1.523.390

Note: To convert from mile to kilometer multiply the value by 1.609.
Source: Eblen 2007

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 region was defined as a 10-mile radial distance (the EPZ) centered on the plant. A sheltering period was defined as the phase occurring before initiation of evacuation procedures. During the sheltering period, shielding factors appropriate for sheltered activity were used to calculate doses to individuals in contaminated areas.

At the end of the sheltering period, residents would begin traveling out of the region. Travel speeds and delay times were based on the evacuation data also found in the *Tennessee Multi-Jurisdictional Radiological Emergency Response Plan for the Watts Bar Nuclear Plant*, Annex H (TVA 2006). General population evacuation times for the various areas within the 10-mile radius were averaged to determine an overall evacuation delay time and evacuation speed. Average evacuation speeds based on the most conservative general population evacuation times in an adverse weather condition were considered (see **Table 6**).

Table 6 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: WBN 2006.

Based on the data cited above, an average evacuation speed of 1 meter per second following a sheltering and evacuation delay time of 45 minutes and 2.50 hours were used. These delay values are provided in the *Tennessee Multi-Jurisdictional Radiological Emergency Response Plan for the Watts Bar Nuclear Plant,* Annex H, (TVA 2006) and NUREG/CR-4551, Vol. 2 (NRC 1990). In addition, consistent with the analysis in the CLWR EIS and the NUREG-1150 evaluation of the Sequoyah Nuclear Plant, it was assumed that 99.5 percent of the population in the 10-mile EPZ would be evacuated.

For this analysis, it was conservatively assumed that persons residing farther than 10 miles away from the plant would continue their normal activities unless the following predicted radiation dose levels were exceeded. At locations where a 50-rem whole body effective dose equivalent in 1 week was predicted, it was assumed that relocation would take place after half a day. If a 25-rem whole body dose equivalent in 1 week were predicted, relocation of individuals in those sectors was assumed to take place after 1 day.

Meteorology

Annual onsite meteorology data sets from 2001 through 2005 were used to prepare the sequential hourly data (8760 hours) required for use in MACCS2 (TVA 2007). The 2002 sequential hourly meteorology data was found to result in the largest risk and was used for all of the analyses presented below. The conditional dose from each of the other years was found to be within 20 percent of the chosen year. The 2003 weather data set was found to result in the lowest population doses. Two sampling methods, bin sampling and stratified random sampling, were used. In bin sampling, the representative subset is selected by sampling the weather sequences after sorting them into weather bins defined by windspeed, atmospheric stability, and intensity and distance of the occurrence of rain. In stratified sampling, the representative subset is selected by randomly sampling hourly weather data from each day. The analysis is based four samples per day. This selection was based on a test by the developer of the MACCS code that indicated that random samples from each 6-hour interval of the year would yield results closer to those obtained from sampling all 8760 hours of the year.

4. Analysis Results

Table 7 summarizes the consequences of the beyond-design-basis accident, with mean meteorological conditions, to the maximally exposed offsite individual and an average individual in the public 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 beyond-design-basis 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 location of the maximally exposed offsite individual may or may not be at the site boundary for these accident sequences because emergency action guidelines would have been implemented and the population would be evacuating from the path of the radiological plume released by the accident. The MACCS2 computer code models the evacuation sequence to estimate the dose to the maximally exposed individual and the general population within 80 kilometers (50 miles) of the accident. **Table 8** summarizes the risks associated with the beyond-design-basis accident to the same receptors in terms of latent cancer fatalities (considering the likelihood of occurrence for each release category). The frequency of each release category is given in Table 4. **Table 9** shows the population dose risks (accident consequence multiplied by the release frequency) for each accident release category.

Table 7 Severe Reactor Accident Consequences

Release Category	*Maximally Exposed Offsite Individual*		*Average Individual Population within 80 Kilometers (50 miles)*	
	Dose (rem)	*Cancer Fatality* [a]	*Dose (rem)*	*Cancer Fatality* [a]
Weather Bin Sampling				
I - Early Containment Failure	64.8	0.078	0.53	0.00032
II - Containment Bypass	15.62	0.0094	0.59	0.00035
III - Late Containment Failure	0.131	0.000079	0.042	0.000025
Weather Stratified Sampling				
I - Early Containment Failure	42.6	0.051	0.43	0.00026
II - Containment Bypass	12.0	0.0072	0.50	0.00030
III - Late Containment Failure	0.153	0.000092	0.037	0.000022

[a] Increased likelihood of cancer fatality based on the health risk factor of 0.0006 cancers per rem for exposures below 20 rem. For exposures greater than or equal to 20 rem, the health risk factor is doubled.

Table 8 Severe Reactor Accident Annual Risks

Release Category	Maximally Exposed Offsite Individual		Average Individual Population within 80 Kilometers (50 miles)	
	Dose [a] (rem/year)	Cancer Fatality [b]	Dose [a] (rem/year)	Cancer Fatality [b]
Weather Bin Sampling				
I - Early Containment Failure	2.2×10^{-5}	2.6×10^{-8}	1.8×10^{-7}	1.1×10^{-10}
II - Containment Bypass	2.2×10^{-5}	1.3×10^{-8}	8.2×10^{-7}	4.9×10^{-10}
III - Late Containment Failure	3.9×10^{-7}	2.4×10^{-10}	1.3×10^{-7}	7.8×10^{-11}
Weather Stratified Sampling				
I - Early Containment Failure	1.4×10^{-5}	1.7×10^{-8}	1.5×10^{-7}	9.0×10^{-11}
II - Containment Bypass	1.7×10^{-5}	1.0×10^{-8}	7.0×10^{-7}	4.2×10^{-10}
III - Late Containment Failure	4.6×10^{-7}	2.8×10^{-10}	1.1×10^{-7}	6.6×10^{-11}

[a] Includes the likelihood of occurrence of each release category.
[b] Increased likelihood of cancer fatality per year.

Table 9 Annual 80-Kilometer (50-mile) Population Dose Risk

Release Category	Weather Bin Sampling	Weather Stratified Sampling
	Dose [a] (person-rem/year)	Dose [a] (person-rem/year)
I - Early Containment Failure	0.28	0.23
II - Containment Bypass	1.25	1.07
III - Late Containment Failure	0.19	0.17

[a] Includes the likelihood of occurrence of each release category. The population within 80 kilometers (50 miles) is projected to be 2,104,700..

The risk results presented here are generally lower than those given in the CLWR EIS for the same accidents due to the use of different data inputs such as lower release frequency, higher core inventory, and slower evacuation speed. The release frequencies used in this analysis are lower by a factor 2 to 5 than those used in the CLWR EIS. These frequencies are based on Revision 3 of the WBN Plant probabilistic safety assessment model (TVA 2007). The ratio of isotopic core inventory is higher by a factor 1 to 3. The evacuation speed is slower by a factor of 2/3, and the population is almost 1.5 times that projected in the CLWR EIS. The latter increases resulted in higher average individual population doses as compared to those projected in the CLWR EIS.

The results presented in Tables 7 through 9 indicate that the highest risk to the maximally exposed offsite individual is one fatality every 38 million years (or 2.6×10^{-8} per year), and highest risk to an average individual member of the public is one fatality every 2 billion years (or 4.9×10^{-10} per year). Overall, the risk results presented above are small. Completion and operation of WBN Unit 2 would not change the risks evaluated here because the likelihood of an accident that could affect both units and lead to radioactive releases beyond those analyzed here is extremely low. 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 initiated by external events. Severe accidents initiated by external events as tornadoes, floods, earthquakes, and fires traditionally have not been discussed in quantitative terms in final environmental statements and were not considered in the GEIS (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.

4.1 Sensitivity Analysis

This section discusses how changes in the analysis assumptions would affect the calculated consequences. The parameters evaluated in this section include release energy, evacuation speed, evacuation fraction with a 16-kilometer EPZ, and release frequency. The effect of the weather sampling method is provided in the baseline analysis above. For the sensitivity evaluations, the input parameters corresponding to Release Category I were used. For each evaluation, only the selected input parameter would be changed.

Release Energy

The release energy (heat content) would lift the plume to a higher elevation where it would spread over a large area downwind from the accident. This effect would reduce the plume contaminant concentration in the vicinity of the plant. Since the analysis used complete washout of the plume at the last ring, the effect of the release energy on the population beyond 50 miles would be negligible. For this analysis, the release energy was reduced from 28 MW to 1 MW. The results indicate that the new population dose risk would decrease by about 22 percent. The dose to nearby residents in the vicinity of the plant would increase, but because these individuals would be evacuated or sheltered, the health effects would be small.

Evacuation Speed

The evacuation speed used in the baseline analysis was 1.0 meter per second, or 2.34 miles per hour. The evacuation time analysis for the 0-to-10-kilometer (0-to-10-mile) area given in the *Tennessee Multi-Jurisdictional Radiological Emergency Response Plan for the Watts Bar Nuclear Plant*, Annex H, (TVA 2006) shows a range of evacuation times from 2 to more than 6 hours, with an average duration value of 4 hours. For the sensitivity analyses, average evacuation speeds of 1.5 and 0.7 meters per second (or 3.36 and 1.57 miles per hour) were used. The new population dose risks for these evacuation speeds were determined to be within 0.96 and 1.06 of the baseline consequences for the cases with the 1.5 and 0.7 meters per second evacuation speeds, respectively.

Evacuation Fraction

The baseline public evacuation fraction within the 16-kilometer (10-mile) EPZ was 99.5 percent. For this analysis, it was assumed that 95 percent of the public would be evacuating. The new population dose risk did not increase the baseline dose risk, the change was within the roundup of MACCS2 numerical output. Therefore, the impact of lower evacuation fraction would be negligible.

Release Frequency

The risks of accidents are proportional to their projected frequency of occurrence. The release frequency values provided in Table 4 are best estimate values. The 95th percentile uncertainty on these estimates could range between 2 to 5 (TVA 2007). Therefore, the population dose risk could vary proportionally as well. The final risk results would be small (see Table 9).

5. MACCS2 Computer Code

The MACCS2 computer code, Version 1.13.1, was used to estimate the radiological doses and health effects that could result from postulated accidental releases of radioactive materials to the atmosphere.

The specification of the release characteristics, designated a "source term," can consist of up to four Gaussian plumes that are often referred to simply as "plumes."

The radioactive materials released are modeled as being dispersed in the atmosphere where they are transported by the prevailing wind. During transport, whether or not there is precipitation, particulate material can be modeled as being deposited on the ground. If contamination levels exceed a user-specified criterion, mitigative actions can be triggered to limit radiation exposures.

Two aspects of the code's structure are basic to understanding its calculations: (1) the calculations are divided into modules and phases, and (2) the region surrounding the facility is divided into a polar-coordinate grid. These concepts are described in the following sections.

MACCS2 is divided into three primary modules: ATMOS, EARLY, and CHRONC. Three phases are defined as the emergency, intermediate, and long-term phases. The relationship among the code's three modules and three phases of exposure are summarized below.

The ATMOS module performs calculations pertaining to atmospheric transport, dispersion, and deposition, as well as the radioactive decay that occurs before release and while the material is in the atmosphere. It utilizes a Gaussian plume model with Pasquill-Gifford dispersion parameters. The phenomena treated include building wake effects, buoyant plume rise, plume dispersion during transport, wet and dry deposition, and radioactive decay and in-growth. The results of the calculations are stored for use by EARLY and CHRONC. In addition to the air and ground concentrations, ATMOS stores information on wind direction, arrival and departure times, and plume dimensions.

The EARLY module models the time period immediately following a radioactive release. This period is commonly referred to as the emergency phase. The emergency phase begins at each successive downwind distance point where the first plume of the release arrives. The duration of the emergency phase is specified by the user and can range between 1 and 7 days. The exposure pathways considered during this period are direct external exposure to radioactive material in the plume (cloudshine), exposure from inhalation of radionuclides in the cloud (cloud inhalation), exposure to radioactive material deposited on the ground (groundshine), inhalation of resuspended material (resuspension inhalation), and skin dose from material deposited on the skin. Mitigative actions that can be specified for the emergency phase include evacuation, sheltering, and dose-dependent relocation.

The CHRONC module performs all of the calculations pertaining to the intermediate and long-term phases. CHRONC calculates the individual health effects that result from both direct exposure to contaminated ground and inhalation of resuspended materials, as well as indirect health effects caused by the consumption of contaminated food and water by individuals who could reside both on and off the computational grid.

The intermediate phase begins at each successive downwind distance point upon conclusion of the emergency phase. The user can configure the calculations with an intermediate phase that has a duration as short as zero or as long as 1 year. Essentially, there is no intermediate phase, and a long-term phase begins immediately upon conclusion of the emergency phase.

These models are implemented on the assumption that the radioactive plume has passed and the only exposure sources (groundshine and resuspension inhalation) are from ground-deposited material. For this reason, MACCS2 requires the total duration of a radioactive release to be limited to no more than 4 days. Potential doses from food and water ingestion during this period are not considered.

The mitigative action model for the intermediate phase is very simple. If the intermediate phase dose criterion is satisfied, the resident population is assumed to be present and subject to radiation exposure from groundshine and resuspension for the entire intermediate phase. If the intermediate phase exposure exceeds the dose criterion, then the population is assumed to be relocated to uncontaminated areas for the entire intermediate phase.

The long-term phase begins at each successive downwind distance point after conclusion of the intermediate phase. The exposure pathways considered during this period are groundshine, resuspension inhalation, and food and water ingestion.

The exposure pathways considered are those resulting from ground-deposited material. A number of protective measures can be modeled in the long-term phase to reduce doses to user-specified levels, including decontamination, temporary interdiction, and condemnation. The decisions on mitigative action in the long-term phase are based on two sets of independent actions: (1) decisions relating to whether land at a specific location and time is suitable for human habitation (habitability), and (2) decisions relating to whether land at a specific location and time is suitable for agricultural production (farmability).

All of the MACCS2 calculations are stored on the basis of a polar-coordinate spatial grid that treats calculations of the emergency phase and calculations of the intermediate and long-term phases somewhat differently. The region potentially affected by a release is represented with an (r,2) grid system centered on the location of the release. The radius, r, represents downwind distance. The angle, 2, is the angular offset from north, going clockwise.

The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid are fixed in the code and correspond to the 16 points of the compass (each is 22.5 degrees wide). The 16 points of the compass are used in the U.S. to express wind direction. The compass sectors are referred to as the coarse grid.

Since emergency phase calculations use dose-response models for early fatalities and early injuries that can be highly nonlinear, these calculations are performed on a finer grid basis than the calculations of the intermediate and long-term phases. For this reason, emergency phase calculations are performed with the 16 compass sectors divided into three, five, or seven equal, angular subdivisions. The subdivided compass sectors are referred to as the fine grid.

The compass sectors are not subdivided into fine subdivisions for the intermediate and long-term phases because these calculations are limited to cancer and genetic effects and do not include estimates of the often highly nonlinear early fatality and early injury health effects. In contrast to the emergency phase, the calculations for these phases are performed using doses averaged over the full 22.5-degree compass sectors of the coarse grid.

Two types of doses, "acute" and "lifetime," may be calculated using the MACCS2 code. Acute doses are calculated to estimate deterministic health effects that can result from high doses delivered at high dose rates. Such conditions may occur in the immediate vicinity of a nuclear power plant following a hypothetical severe accident where containment failure has been assumed to occur. Examples of health effects based on acute doses are early fatality, prodromal vomiting, and hypothyroidism. Lifetime doses are the conventional measure of detriment used for radiological protection. These are 50-year dose commitments to either specific tissues (e.g., red marrow, lungs) or a weighted sum of tissue doses defined by the International Commission on Radiological Protection and referred to as an "effective dose." Lifetime doses may be used to calculate the stochastic health effect risk resulting from exposure to radiation. MACCS2 uses the calculated lifetime dose in cancer risk calculations.

5.1 Data and General Assumptions

To assess the consequences of beyond-design-basis accidents, the following data and assumptions were incorporated into the MACCS2 analysis.

- The **nuclide inventory** at accident initiation (e.g., reactor trip) of those radioactive nuclides that are important for the calculation of offsite consequences is given in Table 2.

- The **atmospheric source term** produced by the accident was described by the number of plume segments released; sensible heat content; timing; duration; height of release for each plume segment; time when offsite officials are warned that an emergency response should be initiated; and for each important radionuclide, the fraction of that radionuclide's inventory released with each plume segment. The release fractions for each accident scenario are provided in Table 3. MACCS2 calculates the atmospheric source terms based on the core nuclide inventory at the time of reactor trip, release time after the reactor trip, and the associated release fractions.

- **Meteorological data** characteristics of the site region were described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at each site. Although 1 year of hourly readings contains 8,760 weather sequences, MACCS2 calculations examine only a representative subset of these sequences. As stated earlier in Section 3.1.2, two types of weather sampling were used: bin sampling and stratified sampling. These two methods are the most used methods selected by MACCS users. Bin sampling requires the user to provide rain intensity at downwind distances; stratified sampling is a purely random selection of hourly data from those occurring at the site. The analysis was based on 139 weather data in bin sampling and 1460 weather data in stratified sampling. Stratified random sampling resulted in less than 10 percent higher doses.

- **Population distribution information** regarding the Watts Bar site was based on data from the 2000 census as used in the SECPOP 2000 computer code (NRC 2003). The generated population data for the site was extrapolated to the year 2030 using the incremental increase in population during the decade recorded from census 1990 to 2000). This data is provided in Table 5 for a polar coordinate grid with 16 angular sectors aligned with the 16 compass directions and 10 radial intervals that extend outward to 80 kilometers (50 miles).

- **Habitable land fractions** for the region around each reactor site were determined in a manner similar to the population distribution. The census block group boundary files include polygons that are classified as water features. The percentage of each sector that is covered by water was determined by fitting this data to the polar coordinate grid.

- **Farmland fractions** are the percentage of land devoted to farming (DOE 1999).

- **Emergency response assumptions** for evacuation, including delay time before evacuation, area evacuated, average evacuation speed, and travel distance, are provided in the Tennessee Multi-Jurisdictional Plans, see Section 3.1.2. Average evacuation speeds are based on the most conservative general population evacuation times.

- **Shielding and exposure data** must be input into the MACCS2 code. The code requires shielding factors to be specified for people evacuating in vehicles (cars, buses); taking shelter in structures (houses, offices, schools); and continuing normal activities either outdoors, in vehicles, or indoors. Because inhalation doses depend on breathing rate, breathing rates must be specified for people who are continuing normal activities, taking shelter, and evacuating. Since indoor concentrations of gasborne radioactive materials are usually substantially less than outdoor concentrations, MACCS2 also requires that inhalation and skin protection shielding factors (indoor/outdoor concentration ratios) be provided.

The protection factors presented in **Table 10** were used in this analysis. The values in this table are for the Sequoyah Nuclear Plant, as stated in NUREG/CR-4551, and were used in the analysis for the WBN Plant.

Table 10 NUREG/CR-4551 Protection Factors

Protection Factor [a]	*Evacuees*	*Sheltering*	*Normal Activities*
Cloudshine Shielding Factor	1.0	0.65	0.75
Skin Protection Factor	1.0	0.33 [b]	0.41 [b]
Inhalation Protection Factor	1.0	0.33 [b]	0.41 [b]
Groundshine Shielding Factor	0.5	0.2	0.33 [c]

[a] A protection factor of 1.0 indicates no protection, while a protection factor of 0.0 indicates 100 percent protection.
[b] These values were based on the recommendation from S. Acharya of NRC as it appears in Appendix A-2 of NUREG/CR-4551, Vol. 2. The recommended values in the report are 0.2 and 0.5 for sheltering and normal activities, respectively (NUREG/CR-4551, Vol. 2, Table 3.12).
[c] This value was based on the recommendation from S. Acharya of NRC as it appears in Appendix A-2 of NUREG/CR-4551, Vol. 2. The recommended value in the report is 0.5 (NUREG/CR-4551, Vol. 2, Table 3.12).

For this analysis, the evacuation and sheltering region was defined as a 10-mile radial distance centered on the plant. A sheltering period was defined as the phase occurring before initiation of the evacuation. During the sheltering period, shielding factors appropriate for sheltered activity were used to calculate doses for individuals in contaminated areas.

At the end of the sheltering period, residents begin traveling out of the region. Travel speeds and delay times are based on the Tennessee Multi-Jurisdictional Plans. The general population evacuation times for the various areas within the 10-mile radius were averaged to determine an overall evacuation delay time and evacuation speed for the WBN Plant.

- Maximally Exposed Offsite Individual (MEI) dose is the total dose estimated to be incurred by a hypothetical individual assumed to reside at a particular location on the spatial grid. Population data, therefore, have no bearing on the generation of this consequence measure. Only direct exposure is considered in these results. Exposures from ingestion of contaminated food and water are not included. In addition, generation of these results takes full account of any mitigative action models activated by exceeding the dose thresholds. During evacuation, individuals have no protection from direct exposure. Therefore, in certain scenarios, it is possible that an evacuee may incur a larger direct exposure dose than an individual who does not evacuate.

- Long-term protective measures such as decontamination, temporary relocation, contaminated crops, milk condemnation, and farmland production prohibition are based on U.S. Environmental Protection Agency (EPA) Protective Action Guides.

- Mitigative actions (relocation, evacuation, interdiction, condemnation) are implemented for beyond-design-basis accidents (vessel breach with containment bypass, vessel breach with early containment failure, and vessel breach with late containment failure).

- Dose conversion factors required by MACCS2 for the calculation of committed effective dose equivalents are cloudshine dose-rate factor; groundshine dose-rate factor; "lifetime" 50-year committed inhalation dose (used for calculation of individual and societal doses and stochastic health effects); and 50-year committed ingestion dose (used for calculation of individual and societal doses and stochastic health effects from food and water ingestion).

5.2 Health Effects Calculations

The health consequences from exposure to radionuclides due to accidental releases were calculated. Total effective dose equivalents were calculated and converted to estimates of cancer fatalities using dose conversion factors recommended by the International Commission on Radiological Protection. For individuals, the estimated probability of a latent cancer fatality occurring was reported for the maximally exposed individual, and an average individual in the population within 80 kilometers (50 miles).

The nominal values of lifetime cancer risk for low dose or low dose rate exposure (less than 20 rad) used in this EIS are 0.0006 per person-rem for a population of all ages, including workers (ISCORS 2002). These dose-to-risk conversion factors are about 20 percent more than those established by the National Council on Radiation Protection and Measurement and used in the CLWR EIS.

The MACCS2 code was applied in a probabilistic manner using a weather bin and a stratified sampling method. Each of the sampled meteorological sequences was applied to each of the 16 sectors (accounting for the frequency of occurrence of the wind blowing in that direction). Individual doses as a function of distance and direction were calculated for each of the meteorological sequence samples. The mean dose values of the sequences were generated for each of the 16 sectors. The highest of these dose values was used for the maximally exposed individual.

6. Conclusions

Table 11 summarizes the consequences of the beyond design-basis accident, with mean meteorological conditions, to the maximally exposed offsite individual, an average individual, and the population residing within an 80-kilometer (50-mile) radius of the reactor site. The analysis assumed that a site emergency would have been declared early in the 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 have been implemented to initiate evacuation of 99.5 percent of the public within 16 kilometers (10 miles) of the plant. The location of the maximally exposed offsite individual may or may not be at the site boundary for these accident sequences because emergency action guidelines would have been implemented and the population would be evacuating from the path of the radiological plume released by the accident.

Table 11 Severe Reactor Accident Annual Risks

Release Category (frequency per reactor year)	*Maximally Exposed Offsite Individual*		*Population within 80 Kilometers (50 miles)*		
			Average Individual		*General public*
	Dose Risk [a] (rem/year)	*Cancer Fatality [b]*	*Dose Risk [a] (rem/year)*	*Cancer Fatality [b]*	*Dose Risk [a] (person-rem/year)*
Maximum dose and risk [c]					
I - Early Containment failure (3.4×10^{-7})	2.2×10^{-5}	2.6×10^{-8}	1.8×10^{-7}	1.1×10^{-10}	0.28
II - Containment Bypass (1.4×10^{-6})	2.2×10^{-5}	1.3×10^{-8}	8.2×10^{-7}	4.9×10^{-10}	1.25
III - Late Containment Failure (3.0×10^{-6})	4.6×10^{-7}	2.8×10^{-10}	1.3×10^{-7}	7.8×10^{-11}	0.19

[a] Includes the likelihood of occurrence of each release category.
[b] Increased likelihood of cancer fatality per year.
[c] These values are taken from Tables 8 and 9; the maximum dose to a maximally exposed offsite individual is from weather bin sampling and the maximum dose to an average individual and population is from weather stratified sampling.

The results presented in this table indicates that the highest risk to the maximally exposed offsite individual is one fatality every 38 million years (or 2.6×10^{-8} per year), and the highest risk to an average individual member of the public is one fatality every 2 billion years (or 4.9×10^{-10} per year). Overall, the risk results presented above are small. Completion and operation of WBN Unit 2 would not change the risks evaluated here because the likelihood of an accident that could affect both units and lead to radioactive releases beyond those analyzed here would be extremely low. This is consistent with the conclusions of NRC's GEIS (NRC 1996). Accidents that could affect multiple units are initiated by external events. Severe accidents initiated by external events as tornadoes, floods, earthquakes, and fires traditionally have not been discussed in quantitative terms in final environmental statements and were not considered in the GEIS (NRC 1996). In the GEIS, however, NRC staff evaluated 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.

7. References

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Enclosure 2

List of Commitments

1. Regarding the status of the following SAMA related design changes (See NRC RAI 5.h.iii) TVA will confirm prior to fuel load:
   - SAMA 65 (install digital feedwater upgrade) – the design change is in process and will be implemented for Unit 2 prior to fuel load.
   - SAMA 198 (improve RHR sump reliability) – the design change has been issued and will be implemented for Unit 2 prior to fuel load.
   - SAMA 271 (refurbish the ERCW pumps and upgrade the capacity of the current pumps) – the design change has been issued and will be implemented for Unit 2 prior to fuel load.
   - SAMA 281 (replace ACAS compressors and dryers) – WBN will refurbish and enhance current units to gain capacity, but not replace units.

2. TVA will confirm prior to fuel load the known differences between the WBN units as shown in response to NRC RAI 1.d.

3. In response to NRC RAI 5.g: SAMA 271 (refurbish ERCW pumps and upgrade capacity of current pumps) – WBN is currently replacing ERCW pumps to support two unit operation. This modification does not change the PRA as modeled and will be implemented into the model in the future through data updates of pump reliability and availability.

4. In response to NRC RAI 5.h.iii: SAMA 218 (improve reliability of power supplies) – the design changes to accomplish this SAMA will be complete by the end of this year except as noted for the batteries mentioned above in SAMA 174.

5. In response to NRC RAI 5.h.iii: SAMA 219 (improve switchyard and transformer reliability) – the design changes to accomplish this item will be completed by the end of the year.

6. In response to NRC RAI 4.b: TVA is evaluating the impact of the SECPOP2000 errors and will provide an update (either revised information using SECPOP2000 Version 3.13.1 or an assessment of the impact on the results) by August 31, 2010.