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

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10 CFR 50.4
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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Units 1 and 2
Facility Operating License Nos. NPF-90 and NPF-96
NRC Docket Nos. 50-390 and 50-391

Subject: **TENNESSEE VALLEY AUTHORITY (TVA) - WATTS BAR NUCLEAR PLANT SEISMIC PROBABILISTIC RISK ASSESSMENT SUPPLEMENTAL INFORMATION**

Reference: TVA letter to NRC, "Seismic Probabilistic Risk Assessment for Watts Bar Nuclear Plant, Units 1 and 2 - Response to NRC Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," dated June 30, 2017 (ML17181A485)

In the Reference letter dated June 30, 2017, TVA provided the Seismic Probabilistic Risk Assessment Summary Report for Watts Bar Nuclear Plant, Units 1 and 2, as requested by NRC's letter dated October 27, 2015 (ML15194A015). The Enclosure to the Reference letter provided the information requested in Item (8)B of the 50.54(f) letter associated with NTTF Recommendation 2.1: Seismic.

The purpose of this letter is to provide supplemental information for the Seismic Probabilistic Risk Assessment Summary Report for Watts Bar Nuclear Plant, Units 1 and 2.

This letter contains no new regulatory commitments.

If you have any questions regarding this submittal, please contact Russell Thompson at (423) 751-2567.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 10th day of April 2018.

Respectfully,

J. W. Shea
Vice President, Nuclear Regulatory Affairs and Support Services

Enclosure: Supplemental Information for the Watts Bar Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic Summary Report

cc (Enclosure):

NRR Director - NRC Headquarters
NRC Regional Administrator - Region II
NRC Project Manager - Watts Bar Nuclear Plant
NRC Senior Resident Inspector - Watts Bar Nuclear Plant

Enclosure
Supplemental Information for the Watts Bar Nuclear Plant, Units 1 and 2, Seismic Probabilistic Risk Assessment in Response to 50.54(f) Letter with Regard to NTTF 2.1 Seismic Summary Report

In response to the 10 CFR 50.54(f) letter issued by the NRC on March 12, 2012 [Ref 1], a Seismic Probabilistic Risk Assessment (SPRA) was developed for Watts Bar Nuclear Plant (WBN), Units 1 and 2. The WBN SPRA was submitted to NRC by letter dated June 30, 2017. The SPRA shows that the point estimate seismic Core Damage Frequency (CDF) is 2.6×10^{-6} per reactor calendar year (rcy) for each unit. The seismic Large Early Release Frequency (LERF) is 1.7×10^{-6}/rcy for each unit. Sensitivity studies were performed to identify critical assumptions, test the sensitivity to quantification parameters and the seismic hazard, and identify potential areas to consider for the reduction of seismic risk. These sensitivity studies demonstrated that the model results were robust to the modeling and assumptions used. No seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights from the seismic risk assessment.

The evaluation of the WBN SPRA relative to the 16 EPRI 1025287 (commonly referred to as the "SPID") [Ref 2] requirement-related items as identified by the "NRC Staff Review Guidance for Seismic PRA Submittal Technical Review Checklist" (ML17041A342), demonstrate that the WBN SPRA is of sufficient quality and level of detail for the response to NTTF 2.1 Seismic.

Below are information summaries associated with the NRC Staff Review Checklist. The information is provided as a guide for the NRC Staff to aid in the location of various pieces of information. References noted are detailed at the end of this enclosure, and are the same as those provided in the WBN SPRA. Specific WBN SPRA Sections are referenced that provide information pertinent to the given topic.

Topic 1: Seismic Hazard (SPID Section 2.1, Section 2.2, and Section 2.3)
Prior to the NTTF 2.1 activities, a probabilistic seismic hazard analysis was initiated to support potential licensing efforts for WBN Unit 2. This analysis [Ref 19] was used for the WBN SPRA in lieu of the NTTF 2.1 submittal [Ref 3] since the site analysis develops the additional elements required for the SPRA, such as Foundation Input Response Spectra (FIRS), hazard-consistent strain-compatible properties, and vertical ground motions. The guidance in the SPID was followed for developing the site analysis. The site analysis is described in WBN SPRA Section 3.1. Figure 3.3-1 of the SPRA compares the mean control point hazard curves at 1 Hz, 10Hz, and PGA for the seismic hazard submitted to the NRC for NTTF 2.1 seismic hazard analysis and the SPRA seismic hazard analysis and shows that, depending on the selected annual frequency of exceedance, the shape of the site analysis is either similar to or slightly above the hazard submitted to the NRC. Peer review findings related to the seismic hazard, except for one, were closed utilizing the process given in Appendix X of NEI 12-13. The open peer review finding, related to screening out of the seismic hazards other than vibratory ground motion, was judged 'Technically Resolved - Open Documentation' by the peer reviewers and thus does not affect the seismic hazard calculation.

Topic 2: Site Seismic Response (SPID Section 2.4)
The site seismic response is provided in Reference 19. The guidance in the SPID was followed for developing the site analysis. The site analysis is described in WBN SPRA Section 3.1. Figure 3.3-2 of the WBN SPRA compares the GMRS submitted to the NRC for NTTF 2.1 with the site response used for the SPRA and shows that the site response is equivalent to or slightly above the response submitted to the NRC for NTTF 2.1. Peer review findings related to the site analysis were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 3: Definition of the Control Point for the SSE-to-GMRS-Comparison Aspect of the Site Analysis (SPID Section 2.4.2)
The GMRS is defined at the Reactor Building foundation control point at a depth of 64 ft. below plant grade of 728 which corresponds to elevation 664 ft mean sea level. Information is provided in Section 3.2 of the WBN SPRA. The site-analysis for the SPRA and the Seismic Hazard Screening Report use the same control point.

Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)
New three-dimensional (3D) finite element models (FEM) were built for the Auxiliary-Control Building (ACB), Diesel Generator Building (DGB), Intake Pumping Station (IPS), and North Steam Valve Room (NSVR). The new FEM, which capture building torsion, out-of-plane floor response, and in-plane floor diaphragm stiffness, satisfy the criteria 1 through 7 in the SPID [Ref 2] Section 6.3.1. The Reactor Building (RB) and the Refueling Water Storage Tank (RWST) use lumped-mass-stick-models. For the Reactor Building, a 3D FEM is used to calibrate the complex interior concrete structure portion of the model. For the refueling water storage tank the existing lumped-mass-stick-model (LMSM) is enhanced. For the Reactor Building and Refueling water storage tank the criteria 1 through 6 in the SPRA Section 6.3.1 is judged to be met. Information is provided in Section 4.3.3 of the WBN SPRA. Peer review findings related to the adequacy of structural modeling were closed utilizing the process given in Appendix X of NEI 12-13.

The 3D models developed incorporated the geometry, configuration, and dimensions of the structural components of the building, such as the foundation and floor slabs, walls, and openings with reference to the respective mid-planes. The models automatically incorporate coupling between horizontal directions and also coupling between vertical and horizontal directions. With the exception of the DGB, which was reanalyzed after peer team review using one combined structural model, the other buildings are analyzed using two different models in the horizontal and vertical directions. Horizontal models assume all concrete sections are cracked, whereas vertical models assume only floor and roof concrete elements are cracked. Since 3D FEM analyses for these buildings reveal that the coupling is not significant (especially between the horizontal and the vertical directions), the combined SSI result from the horizontal and vertical models effectively reduces the out-of-plane bending and in-plane shear stiffness to about 50 percent, and preserves the axial stiffness to uncracked stiffness. As stated above, the DGB was reanalyzed after peer review using one combined structural model.

The RB and the RWST use the LMSMs reported in the FSAR. The RB includes three structural systems, Interior Concrete Structure (ICS), Steel Containment Vessel (SCV), and Shield Building (SB) supported by a common foundation mat. The use of LMSMs for the RWST, SB and the SCV is justified on the basis that these structures are relatively simple and symmetric. The LMSM for the ICS is verified and validated by means of independent evaluations using FEMs. Both the horizontal and vertical models of the ICS are configured to capture the coupling between vertical and horizontal directions.

Criteria 4 in SPID Section 6.3.1 states that the number of nodal or dynamic degrees of freedom should be sufficient to represent significant structural modes. Cutoff frequencies are 50 Hz for the ACB, RB, and NSVR, 90 Hz for the IPS and RSWT, and 100 Hz for the DGB. Structural finite element models are sufficiently discretized to capture modes higher than 20 Hz in all directions.

Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

The NSVR uses a fixed-based dynamic seismic analysis. Because the NSVR is a relatively small structure located on the side of the much larger RB, the seismic analysis of the NSVR assumes fixed-base foundation conditions subjected to the seismic motion of the RB at elevations below grade. The analysis method is described in Section 4.3.1 of the WBN SPRA. Peer review findings related to the adequacy of structural modeling were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Scaling of In-Structure-Response-Spectra (ISRS) to account for higher ground motion levels was not used for the WBN SPRA.

The fragilities of five nuclear steam supply system (NSSS) components, Reactor Vessel, NSSS Piping, Steam Generator, Reactor Coolant Pump, and Pressurizer, are based on scaling the reported components' stresses, displacements, and support reactions. The scaling is based on the previously developed in-structure response spectra (SSE spectra), NSSS natural frequencies, mode shapes, and participation factors. Review level earthquake (RLE) seismic demand is obtained by scaling the reported stress due to the SSE by the ratio of the spectral accelerations at appropriate frequencies represented in the SSE spectra and the in-structure response spectra due to the RLE. Accordingly, seismic safety factors for NSSS components are calculated using the scaled demand and the capacity. Then, appropriate inelastic absorption factors are utilized to incorporate non-linear effects.

Topic 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

The requirements for new analysis are found in the ASME/ANS SPRA [Ref 4] standard under high level requirement SFR-C. The site response is developed with appropriate building specific soil profiles that captures the uncertainty and variability in material dynamic properties as described in Sections 3.0 and 4.3 of the WBN SPRA. Peer review findings related to the adequacy of new response analysis were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)
The approach to screening out rugged SSCs from the model followed the guidance in Section 6.4.3 of the SPID. Other component capacity screening was not performed for WBN because the WBN GMRS seismic hazard is considerably higher than the available tools for seismic capacity screening. The screening approach is documented in the WBN SPRA Sections 4.2 and 4.3.6. Peer review findings related to the adequacy of selection of SSCs for seismic fragility analysis were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)
The CDFM/Hybrid methodology used for fragility analysis is documented in Section 4.3.7 of the WBN SPRA and meets the recommendations in Section 6.4.1 of the SPID [Ref 2]. Recommended values from Table 6-2 of the SPID [Ref 2] were used to develop full seismic fragility curves. Peer review findings related to the adequacy of CDFM/Hybrid Methodology for fragility analysis response analysis were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)
Devices of the type identified in EPRI Phase 2 testing (EPRI 3002002997, *High Frequency Program, High Frequency Testing Summary, September 2014*) as being potentially sensitive to high-frequency seismic motion were included and documented in Sections 4.1.1 and 4.1.2 of the WBN SPRA. Peer review findings related to capacities of SSCs sensitive to high-frequencies were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)
Circuit analysis was relied on for screening relays and other components potentially sensitive to high-frequency vibratory motion. Circuit analysis was performed to identify relays that can potentially impact plant SSCs if chatter were to occur, and screen out the relay devices that do not pose a safety concern. The circuit analysis was performed in accordance with the requirements in the ASME/ANS SPRA Standard [Ref 4] and meets the SPID [Ref 2] requirements, and is documented in Section 4.1.1 of the WBN SPRA. The circuit analysis resulted in most relay chatter scenarios screened from further evaluation based on no impact to component function. However, some relays did not screen from further evaluation. Further fragility calculations and evaluations were performed for the relays that did not screen. The uncertainty parameters (βr and βu) for the relays were based on SPID Table 6-2. Uncertainty parameters different from the recommended values in the SPID were not used. The further calculations and evaluations showed that the unscreened relays were acceptable because either the relay median ground acceleration (A_m) exceeded the fragility cutoff used in the SPRA model or the relay is in a system was not credited in the SPRA model. No operator actions were used to resolve relay chatter. Peer review findings related to capacities of relays sensitive to high-frequencies were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

The CDFM methodology has been used in the seismic PRA for analysis of the bulk of the SSCs requiring seismic fragility analysis. Selected SSCs that are significant contributors to the seismic risk were evaluated using separation of variables approach. This is consistent with the requirement in SPID Section 6.4.1 [Ref 2] and documented in the WBN SPRA Section 4.3.8.

To account for differences between test response and required response spectra bandwidths in determination of component fragilities, both the test response spectra and required response spectra for a tested component are clipped in accordance with the methodology presented in Appendix Q of EPRI-NP-6041. The clipping is done prior to any comparisons of the spectra to ensure that only broad band comparisons are made when determining the limiting TRS/RRS ratio.

To account for appropriate reduction of test levels performed using single- or dual-axis testing in the determination of component fragilities, F_{MS} factors from EPRI-NP-6041 App. Q (page Q-9) are used. Note that most of the tests are multi-axis but when single axis or dual axis testing is completed, the appropriate factor is used.

To account for multi-mode response in determination of component fragilities for components that were tested using narrowband excitation, component TRS/RRS ratios are determined over a frequency range of interest where damage to the component is expected to occur, usually at the dominant mode of the component. In instances where damage may be expected to occur at higher modes, the TRS/RRS ratio is calculated as the minimum ratio over the frequency range of interest.

In general, the fragilities consider seismic motion in all three directions. Each component is reviewed and if it is determined that a particular direction would be non-damaging to a component, then the fragility analysis focused on the directions of motion leading to potential damage. This is done to be as realistic as possible. As an example, Motor Control Centers are commonly installed in long rows within a building. Since all cabinets are bolted together, then the side to side natural frequency is much higher than the front to back (and much higher than the side to side natural frequency in a test report that investigated only one cabinet section). This natural frequency in the side-to-side direction is well above 20 Hz, so the seismic displacements are insignificant and judged to be non-damaging for component functionality (except that sensitive relays are specifically evaluated). In this scenario, TRS/RRS ratios are only investigated in the front to back and vertical directions. The minimum ratio in either of those two directions over the frequency range of interest is used to calculate the fragility.

The method used to select the dominant risk contributors on which SoV fragility calculations were performed as described below.

A component is designated as a significant contributor to the model if it has a Fussel Vesely value greater than 0.005. On an iterative basis, fragility components were refined until a subsequent iteration of the quantification showed no significant change in the number of components or rankings of the components on the top contributor list. The tables in the SPRA submittal are the result of multiple iterations.

For all WBN building structures, the CDFM method was used for fragility calculations. For all components aside from those within the DGB, the CDFM methodology was used. A sensitivity study was performed that demonstrates, based on the soil and site conditions of the plant, that there is little difference in component fragility between the results of the detailed CDFM fragility calculations performed for WBN components, and the SoV Method. The sensitivity study investigated a top contributor relay logic panel, a representative switchgear cabinet, and a top contributor block wall. The sensitivity study concluded that the detailed CDFM fragility calculation method used at WBN resulted in realistic fragility parameters.

After the second-to-last quantification, more detailed analysis was performed for the DGB and all of its components, because (1) many of the components in the DGB were showing up as top contributors, (2) the WBN hazard range of interest was determined to be above the hazard range considered for the strain compatible soil properties used for SSI analysis of the DGB, and (3) the peer review team noted some anomalies in a DGB block wall calculation. The results of the DGB re-analysis using the SoV method showed that the capacities for all dominant contributors from the second-to-last quantification remained the same or slightly increased, with one exception being for N/S shear wall capacities, which decreased by up to 20%. However, these shear walls have such high capacity that the 20% strength decrease was not significant.

Prior to the final quantification, Flexible and Diverse Coping Strategies (FLEX) equipment was only evaluated through a sensitivity study. For F&O resolution, FLEX equipment was included in the model for the final quantification. The FLEX component fragilities were calculated using margin assessment from the Expedited Seismic Evaluation Process (ESEP) and therefore are conservative. To determine the significance of the conservative FLEX component fragilities, a sensitivity study was completed increasing FLEX fragility values by 20%. The results demonstrated that increase of these fragilities did not make a significant change to SCDF or SLERF (2% and 1% reductions, respectively) and support the conclusion that current fragility calculations are sufficient for these items.

In the final quantification, several new items came up to the top contributor list for SCDF and SFERF, and no explicit SoV analyses were performed for the majority of these items. Based on the level of rigor in the CDFM fragility calculations performed for WBN, and the results of the SoV sensitivity studies and DGB re-investigation, it is concluded that the current fragility calculations are sufficiently realistic and representative of as-installed plant conditions.

Topic 13: Evaluation of LERF (SPID Section 6.5.1)
The evaluation of LERF is judged to be consistent with section 6.5.1 and Table 6-3 of the SPID [Ref 2]. Sections 5.1 and 5.5 of the WBN SPRA detail the evaluation of LERF. Peer review findings related to the evaluation of LERF were closed utilizing the process given in Appendix X of NEI 12-13.

Topic 14: Peer Review of the Seismic PRA, Accounting for NEI 12-13 (SPID Section 6.7)
The peer review of the seismic PRA meets the elements described in Section 6.7 of the SPID [Ref 2]. Peer review findings, except for one, were closed utilizing the process given in Appendix X of NEI 12-13. The open peer review finding related to screening out seismic hazards other than vibratory ground motion was judged 'Technically Resolved - Open Documentation' by the peer reviewers and thus is not significant to the SPRA conclusions for this review application. Information is provided in Appendix A of the WBN SPRA. ASME / ANS-Sb-2013 was used by the SPRA peer review team and the independent assessment team.

Many of the internal events F&Os provided in Appendix A of the submittal have been subsequently closed through the closure review process, including 1-8, 3-1, 3-8, 5-1, 7-4, and 7-8. The closure review was performed in accordance with the process documented in Appendix X to NEI 05-04, as well as the requirements published in the ASME/ANS PRA Standard (RA-Sa-2009) and Regulatory Guide 1.200, Revision 2. The model that was reviewed in the closure review is the same model that was used as the basis for the WBN SPRA. Therefore, these F&Os have no impact on the WBN SPRA.

Internal events F&O 3-6 remains open because the state of knowledge (SOK) correlation was not applied in the ISLOCA calculation of valve failure probabilities (as required by QU-A3). However, the SPRA model is not affected since similar components are grouped fragility groups that are completely correlated. In addition, common cause grouping of random failures for similar components is included WBN SPRA, which accounts for a majority of the effects of the SOK Correlation. Therefore, there is no impact on the WBN SPRA results.

No PRA upgrades, as defined per PRA Standard ASME/ANS-Sa-2009, were made between the time of the 2009 internal events PRA peer review and the version of the internal events PRA model which was used to develop the subsequent SPRA model. The table in Appendix A of the WBN SPRA states that for each F&O, there is no upgrade and no new method used to resolve the F&O. Therefore, a follow-on peer review was not required.

NEI 05-04/07-12/12-06 Appendix X describes the process by which an independent assessment team can close out F&Os. The NRC staff accepted this process, as modified by two conditions, by letter dated May 3, 2017 (ML17079A427). The independent assessment review team adhered to the requirements of Appendix X and the conditions in the NRC letter. The independent assessment team reviewed the documented finding closure basis prepared by TVA, including updated PRA models and documentation. Appendix C of the Finding Closure Review report provides a table that summarizes both TVA's input to the review team and the review team's assessment of adequacy of closure. This table includes a column labeled "Independent Review Team Assessment" in which the review team's basis for determining whether each finding was adequately addressed is stated, with references to the revised PRA documentation as appropriate. The same process was used to assess adequacy of resolution of each finding within the review team's scope, including the following findings associated with SRs SFR-A2, SFR-F1, and SFR-G2. SFR-A2 associated findings 23-5, 23-8, 24-15. SFR-F1 associated findings 22-6, 22-7, 23-14, 23-15, 23-21, 23-22, 24-10, 24-12, 24-14. SFR-G2 associated findings 23-3, 23-6, 23-7, 23-9, 23-11, 24-4, 24-7, 24-11, 24-13. The independent assessment team members with responsibility for fragility review conducted their review and agreed with plant response on the associated findings for SFR-A2, SFR-F1, and SFR-G2.

Section 2.2.5 of the Finding Closure Review report notes that TVA did not identify any of the finding dispositions as representing PRA Upgrade, i.e., all were identified as PRA Maintenance, and further confirms that the review team did not identify any of the finding closure dispositions as upgrades.

Topic 15: Documentation of the Seismic PRA (SPID Section 6.8)
WBN SPRA Table 2.0-1 provides a cross-reference for 50.54(f) Enclosure SPRA Reporting and Table 2.0-2 provides a cross-reference for additional SPID Section 6.8 Reporting. The tables show how the documentation requirements of Section 6.8 of the SPID [Ref 2] are met. Appendix A of the WBN SPRA provides the summary of the SPRA Peer Review and Assessment of the PRA Technical Adequacy.

Topic 16: Review of Plant Modifications and Licensee Actions, If Any
There are no modifications necessary to achieve the appropriate risk profile as documented in Section 6 of the WBN SPRA. No seismic hazard vulnerabilities were identified, and no plant actions have been taken or are planned given the insights from the seismic risk assessment.

TVA reviewed the risk levels and found the risk level to be low and tolerable with respect to total plant risk, therefore no modifications were considered. Prior to SPRA development and in conjunction with the Expedited Seismic Evaluation Program, a program to evaluate the seismic margin was initiated. The purpose of the program was to conduct thorough review of the seismic IPEEE for possible increase in the seismic margin from a 0.30g review level earthquake (RLE) to a HCLPF capacity of at least 0.50g RLE. These efforts included reviewing possible modifications to increase capacities. In addition, the IPEEE was updated from a focused scope to a full-scope IPEEE. Through these efforts, Watts Bar improved the plant HCLPF seismic capacity to 0.50g RLE. Anchorage modifications were made to the 480V Shutdown Board Transformers to improve the HCLPF from 0.38g to 0.85g. No additional modifications were found to be required under this review. No single vulnerability has been shown to significantly impact CDF or LERF results, with the exception of the assumed fragility for Loss of Offsite Power. Therefore, no plant modifications were considered following the issuance of the WBN SPRA model. Any refinements in the fragility values of the top contributors to risk are not expected to significantly affect overall seismic risk. In addition, any improvements in the fragilities of the top contributors are likely to affect both CDF and LERF, so the ratio of CDF to LERF is expected to be the same.

For seismically induced failure of human actions due to instrumentation failures (SEIS_HRAINSTR), the individual instrumentation components were not explicitly modeled in the WBN SPRA. Instead, a bounding event, SEIS_HRAINSTR, was modeled with operator actions that represented failure of all instrumentation, simultaneously. The bounding instrument fragility was conservatively calculated considering the types of racks and panels on which they are mounted. Additionally, the control room ceiling and instrumentation power supplies were also considered in the calculation. Improving the fragility of one or more Auxiliary Instrument Room panels would not improve the overall instrument failure probability unless all panels were modified. Additionally, if the A_m value for all panels were improved to greater than or equal to 2.08g, the main control room (MCR) ceiling becomes the bounding fragility. Similarly, improving the fragility of the MCR 120 Power Supply Distribution Panels will improve the overall instrument failure probability, up until it is equal to or greater than 2.09g which is the fragility of the MCR 120V Instrument Power Transformers.

Seismically induced internal flooding (SEIS-IF), the fragility represents a failure of the bracing for the piping, and effectively represents failure of any piping brace that can lead to a modeled internal flooding event throughout the plant. This is a collection of several uncorrelated fragility events conservatively modeled to initiate all internal flooding events. If the bracing was improved at one or more locations, this event would have the same effect unless all pipe bracing was improved. Furthermore, the other failure modes or failures of other equipment would likely increase in importance if all pipe bracing was improved.

Breaker chatter for low voltage switchgear (LVS) and breaker chatter of medium voltage switchgear (MVS) are represented by SEIS_0-24 and SEIS_0-25, respectively. The failure of these events conservatively represents failure of all breakers in each group, respectively. The fragility calculations for these breakers have been highly refined and further evaluation is not expected to have a significant ability to reduce risk. It should be noted that these fragility groups predominantly contribute to CDF and LERF in the higher seismic bins (>1.2g) and further refinement would likely result in higher importance of other fragility groups with similar fragility values. No modifications that would significantly improve breaker fragilities have been identified.

From the review of the plant seismic margin prior to SPRA development and the review of the SPRA risk levels and top contributors, potential modifications that are easily completed to improve equipment capacities have been completed and no additional modifications were found to significantly improve plant seismic risk levels.

Sensitivities were performed on the SPRA results. Conservatisms and non-conservatisms in the model exist and are discussed as follows:

1. Complete seismic correlation of most fragility groups is a conservatism. However, a sensitivity analysis performed for this item revealed a maximum of a 16.8% reduction in SLERF if all groups are completely uncorrelated, and complete correlation is expected to be much closer to the actual result. Therefore, the level of conservatism introduced by excessive correlation is not deemed to be a significant factor.

2. The modeling of seismic impacts on evacuation may be conservative, given that all large seismic releases were treated as SLERF for events >0.5g. This effectively assumes that for all events greater than 0.5g, no evacuation of the surrounding area is possible. However, a sensitivity study identified the maximum possible contribution from the treatment to be 5.1%, and since large seismic events are expected to significantly impact evacuation capabilities, the resulting skew in SLERF should again be significantly lower.

3. A fragility cutoff value of 3.5g was used, a potentially non-conservative approach. Thus, anything that had an A_m>3.5g was screened from inclusion in the model, with the exception of some events directly tied to SIET (seismic initiating event tree) events. Sensitivity showed a maximum contribution to SLERF from events greater than this of 2%, by introducing a direct core damage event at this fragility.

4. The lowest applicable failure mode fragility for a building was conservatively assumed to fail all components within that building. The only structure included in the SLERF importances was the NSVR, which was split into an A state and a C state to partially address this concern. This change resulted in a FV of the A mode failure of the NSVR of 0.008, which is above the screening limit of 0.005, but still represents less than a 1% contribution to SLERF. In addition, though the turbine building did not appear on the importances directly, HFEs in the turbine building were considered failed due to inaccessibility, which failed FLTB1C and OPCMPA for all but bin 1. Note that this does not necessarily assume a guaranteed failure of SCCW piping leading to major flooding conditions in the turbine building due to the building's collapse, a scenario which was analyzed to increase SLERF by 0.7% in another sensitivity study.

5. In general, components inside non-Seismic Category 1 structures are assumed to fail. This is conservative, but a sensitivity study (which included all equipment in non-cat 1 structures, except for the SCCW piping failure described above in point 4) shows no significant impact on the model due to this assumption.

6. The PCS is assumed to fail. This approach is conservative and the sensitivity is examined in the SQU Notebook.

7. Seismically-rugged components (check valves, manual valves, strainers, and filters) are assumed to not fail for the seismic events modeled. Given that the worst case fragility values for these components were above the fragility cutoff mentioned in point 3 above, this does not introduce further non-realism.

8. Assuming complete seismic correlation for ruptures in the Main Steam lines most likely results in those events resulting in direct core damage rather than being transferred to the SSBI and SSBO event trees. Modeling this event in the SSBI and SSBO event trees (which do not automatically lead to core damage) is potentially non-conservative. Sensitivity study performed showed essentially no impact due to this modeling.

9. The baseline SPRA modeling generally does not include > 24-h mission potential impacts such as additional room cooling, coolant makeup, steam supply, and 72-h mission for failure to run. This approach is non-conservative with respect to the SPRA base case results. Sensitivity study indicated a potential increase in SLERF of 0.7% due to this change.

10. Seismically induced failure of the control room ceiling (as a bounding condition to control room inoperability) in conjunction with failure of the operators to shut down the plant remotely is assumed to lead to direct SLERF. This is a conservative assumption, but the CR ceiling event does not appear in the SLERF importances.

11. Sub-components that were mounted on other SEL equipment were considered "boxed" by the larger component, e.g. control switches boxed on a board or limit switches boxed on a valve. In cases where all components can contribute to the overall loss of the larger component, only the governing fragility was used. This excludes chatter sensitive components (relays). Any chatter sensitive component is analyzed separately, independent of the larger component and will have a separate event in the SPRA model.

12. EPRI HRA Bin 4 (seismic bins 7 and 8) assume that all operator actions are unsuccessful due mainly to lack of HRA instrumentation. This includes seismic events in excess of 2.0g. This may potentially have the effect of raising HFE FVs and lower HFE RAWs, but the potential impact of this event is very minimal given the other large probabilities throughout these bins.
13. HFEs involving main feedwater and fire water alignment were set to 1.0 for all bins, since those systems were set to guaranteed failure in the model (see point 5 above).
14. For the relay chatter analysis, chatter was assumed to create the most undesirable combination of individual contact pair chattering. Given that no relay chatter events were ultimately included in the SPRA, this did not impact the SLERF results.

The approach for determining the importance measures in the SPRA in the context of the "binning" is discussed as follows. Per the standard, seismic events were split into 8 bins, each with a representative fragility. For importances, the resulting cutsets were modified from CCDP/CLERP and into CDF/LERF cutsets, merged, and then analyzed using the ACUBE importances tool. There is no expected non-realism introduced into any of these importance measures with the exception of the HRA RAW results. Due to the HRA combination method, it is extremely difficult to develop a precise number. Therefore, the HRA RAW events were generated using a conservative process. Specifically, all HRA combo and recovery events that related to an HFE were set to TRUE (rather than just converted to the corresponding combo event that did not contain the HFE). Since this did not result in any risk-significant RAW HFEs, this method was not refined.

<u>References</u> (as listed in WBN SPRA Section 7.0)

1) NRC (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., “Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident,” March 12, 2012, ML12056A046.

2) EPRI 1025287, Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic. Electric Power Research Institute, Palo Alto, CA: February 2013.

3) TVA, (Shea) Letter to NRC, “Tennessee Valley Authority’s Seismic Hazard and Screening Report (CEUS Sites), Response to NRC Request for Information Pursuant to 10CFR50.54(f) Regarding Recommendation 2.1 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident”, March 31, 2014, ML14098A478.

4) ASME/ANS RA-S-2008, *Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, including Addenda *B, 2013*, American Society of Mechanical Engineers, New York, *September 30, 2013.*

19) “Seismic Hazard Analysis - Seismic Probabilistic Risk Assessment”, TVA Calculation CDN000002015000739, Rev 2.