UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

July 23, 2013

MEMORANDUM TO: Memo To File

FROM: Scott Shaeffer, Chief ***/Eugene Guthrie RA for/***
Projects Branch 6
Division of Reactor Projects

SUBJECT: SUBMITTAL OF REFERENCE DOCUMENTS RELATED TO EA-13-0118 FOR BROWNS FERRY NUCLEAR PLANT UNIT 2, DOCKET 50-260

This memo submits reference documents reviewed by the NRC that are to be made available for public viewing. The enclosed documents are as follows:

Enclosure 1: TVA Memorandum on a Regulatory Analysis Performed for Browns Ferry Unit 2 Performance Deficiency (July 19, 2013)

Enclosure 2: PRA Evaluation Response BFN-2-13-096 (July 18, 2013)

Docket No.: 50-260
License No.: DPR-53

Enclosures: As Stated

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ADAMS: ☐ Yes ACCESSION NUMBER:____ ___________ ☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

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July 19, 2013

Joe Shea, LP 3D-C

MEMORANDUM ON A REGULATORY ANALYSIS PERFORMED FOR BROWNS FERRY UNIT 2 PERFORMANCE DEFICIENCY

The purpose of this memorandum is to forward a regulatory analysis that evaluated a performance deficiency at Browns Ferry Unit 2. The performance deficiency was the failure to follow an Operating Instruction (OI). This regulatory analysis concludes the following. The failure to follow OI 2-OI-99 is a performance deficiency. This performance deficiency is more than minor. No identification credit is warranted since the performance deficiency was self revealing. If the performance deficiency is taken in isolation, then it should have screened to "green" without the need to perform a detailed significance determination. However, if plant conditions at the time the performance deficiency occurred are considered, then the standard would be met and the performance deficiency would require a detailed risk analysis. Using the change in core damage frequency (ΔCDF) and the change in large early release frequency (ΔLERF) as the figures of merit for the determination of significance, the calculated values for both ΔCDF and ΔLERF meet the criteria to be classified as "green." The TVA's calculation of ΔCDF and ΔLERF is conservative. Please see the attachment for the complete analysis.

This analysis has been coordinated with Nuclear Power Group Probabilistic Risk Analysis staff and management and Brown's Ferry Nuclear Plant Site Licensing staff and management.

P. R. Wilson
Site Licensing Oversight Manager
NPG Nuclear Licensing
LP 4B-C

PRW:
cc (Attachment): Regulatory Analysis of Failure to Follow an Operating Instruction
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Regulatory Analysis of Failure to Follow an Operating Instruction

The following is a regulatory analysis of a performance deficiency that occurred at Browns Ferry Unit 2 on December 22, 2012. This analysis includes a description of the performance deficiency, an evaluation of identification credit, an issue screening review and analysis of the TVA's significance determination that was performed to assess the change in risk due to the performance deficiency. This analysis used the guidance in the NRC's Inspection Manual Chapters as the standard for this evaluation.

Description of the Performance Deficiency

On December 22, 2012, a licensed operator failed to follow a step in Operating Instruction (OI) 2-OI-99, "Reactor Protection System" and inadvertently opened the 2A Reactor Protection System (RPS) Motor Generator Breaker. This resulted in the de-energization of the 2A RPS Bus. The loss of power to this bus resulted in a half reactor scram[1] . At the time that this performance deficiency occurred, the 2B RPS bus[2] was also de-energized due an unrelated malfunction that occurred during emergency diesel testing. The loss of power to both RPS busses resulted in a Unit 2 reactor scram and the closure of the Main Steam Isolation Valves (MSIVs). The Reactor Core Isolation Cooling System (RCIC) and the High Pressure Coolant Injection System (HPCI) automatically initiated as designed. Reactor pressure control was established by manually operating the Safety Relief Valves and water level control was established with RCIC system. The HPCI system was returned to standby readiness. Approximately four hours after the scram, MSIVs were opened, and the Main Condenser was re-established as the primary heat sink.

Performance Deficiency Standard

NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," defines a performance deficiency as "an issue that is the result of a licensee not meeting a requirement or self-imposed standard where the cause was reasonably within the licensee's ability to foresee and correct, and therefore should have been prevented." The failure to follow 2-OI-99 meets the NRC definition of a performance deficiency since 2-OI-99 is a procedure required by Unit 2 Technical Specification 5.4.1, Procedures.

Identification Credit

An analysis was performed to determine if the TVA could be credited for identifying the performance deficiency. In IMC 0610, performance deficiency identification can be NRC identified, self revealing or licensee identified. As defined in IMC 0612, "self-revealing findings

[1] The de-energization of the 2A RPS Bus also results in the following: Primary Containment Isolation System (PCIS) Group 1 half-trip logic de-energized; a PCIS Group 2 isolation; a PCIS Group 3 isolation; a PCIS Group 6 isolation, a PCIS Group 8 isolation; Control Room Emergency Ventilation System start; and a Standby Gas Treatment System start.

[2] Aside from a half scram signal resulting from the loss of power to the 2B RPS bus, the loss of power also resulted the following; Primary Containment Isolation System (PCIS) groups 2,3,6 and 8 isolations and the automatic initiation of the associated train of Standby Gas Treatment and Control Room Emergency Ventilation.

or violations are those developed from issues that become self-evident and require no active and deliberate observation by the licensee or NRC inspectors to determine whether a change in process or equipment capability or function has occurred. Self-revealing issues become readily apparent to either NRC or licensee personnel through a readily detectable degradation in the material condition, capability, or functionality of equipment or plant operations and require minimal analysis to detect." This performance deficiency meets the definition of a self revealing finding and no credit can be given as a licensee identified finding.

Issue Screening Analysis

An analysis was performed to assess if the performance deficiency should be consider minor or more than minor. NRC IMC 0612 Appendix B, "Issue Screening" provides the following guidance.

"If the answer to any of the following questions is "yes," then the performance deficiency is More-than-Minor and is a finding. If the answer to all of the following questions is "no," then the performance deficiency is minor and is not a finding.

- Could the performance deficiency reasonably be viewed as a precursor to a significant event?
- If left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern?
- Does the performance deficiency relate to a performance indicator that would have caused the performance indicator to exceed a threshold?
- Is the performance deficiency associated with one of the cornerstone attributes listed at the end of this attachment and did the performance deficiency adversely affect the associated cornerstone objective?"

The failure to follow 2-OI-99 can reasonably be viewed as a precursor to a significant event. The event in this case is a reactor scram with the loss of the primary heat sink. Therefore, this performance deficiency is more than minor.

Next, this analysis reviewed whether the performance deficiency could be screened to "green." The NRC's screening guidance is contained in IMC 0609 Appendix A, "The Significance Determination Process (SDP) for Findings at Power." Since the performance deficiency resulted in the generation of a half scram, the performance deficiency can conservatively be characterized as a transient initiator. The screening question in IMC 0609 Appendix A for transient initiators is as follows. "Did the finding cause a reactor trip AND the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g. loss of condenser, loss of feedwater)?" This performance deficiency, taken in isolation, would not meet this criterion and should screen as "green." However, if plant conditions at the time the performance deficiency occurred are considered, then the standard would be met and the performance deficiency would require a detailed risk analysis.

Significance Determination Regulatory Assessment

The TVA performed a detailed risk evaluation of this performance deficiency. The following is a summary and regulatory assessment of this analysis.

The NRC's IMC 0308 Attachment 3, "Significance Determination Basis Document," states the following. "The additional annualized core damage frequency (CDF) risk due to deficient licensee performance must be dependent only upon the performance issue itself and not the particular plant configurations during which the issue occurred. Therefore, if a degraded equipment or function is identified to exist simultaneously with equipment outages for preventive maintenance or testing, the SDP inputs cannot include the contribution of the maintenance or testing, since this is already included in the normal annualized CDF against which the change is being measured." This suggests that the reactor scram and MSIV closure does not need to be considered in the TVA's risk analysis. However, to be conservative, the scram and MSIV closure were modeled in the TVA's risk analysis.

The TVA's risk evaluation assessed the significance of the performance deficiency in terms of the resulting change in core damage frequency (ΔCDF) and change in the large early release frequency (ΔLERF). IMC 0609, "Significance Determination Process," defines ΔCDF and ΔLERF as figures of merit for significance determinations and are the figures of merit most commonly used by the NRC. IMC 0609 characterizes the significance of performance deficiencies as follows.

- Red (high safety or security significance) is quantitatively greater than 10^{-4} ΔCDF or 10^{-5} ΔLERF.

- Yellow (substantial safety or security significance) is quantitatively greater than 10^{-5} and less than or equal to 10^{-4} ΔCDF or greater than 10^{-6} and less than or equal to 10^{-5} ΔLERF.

- White (low to moderate safety or security significance) is quantitatively greater than 10^{-6} and less than or equal to 10^{-5} ΔCDF or greater than 10^{-7} and less than or equal to 10^{-6} ΔLERF.

- Green (very low safety or security significance) is quantitatively less than or equal to 10^{-6} ΔCDF or 10^{-7} ΔLERF.

For this performance deficiency, the ΔCDF is calculated by multiplying the change in initiating event frequency (IE) and conditional core damage probability (CCDP). The CCDP represents the margin to core damage given the occurrence of an initiating event and is basically a characterization of the ability of the plant to cope with an event. The change in the IE is the increase in the frequency of an initiating event in terms of occurrences per year or 1/N where N equals number of years between occurrences.

The CCDP for this performance deficiency was calculated to be 3.6E-6. This calculation assumed the performance deficiency caused the reactor scram and a loss of the main condenser (surrogate for MSIV closure). The calculation also assumed some credit for the operators to reopen the MSIVs for longer term sequences where HPCI and RCIC do not fail early. The calculated CCDP for this performance deficiency is very similar to that calculated by the NRC in Inspection Report 05000260/2013012.

The change in IE frequency due to the performance deficiency was calculated to be 0.15/Yr. This IE was derived based on the following considerations. A review of the BFN Operations Narrative Logs for the past five years found 66 instances of transferring RPS using OI 2(1,3)-01-99 across the three units. This equates to approximately 13 transfers per year. All of these transfers were completed successfully with the exception of the December 22, 2012 transfer. Using this data directly to estimate the probability of failure to successfully execute the procedure would result in a probability of 1/66. However, this would not be the frequency of a scram. However, it does illustrate that the failure to follow the procedure correctly is relatively low.

The change in IE frequency is therefore evaluated as follows. The performance deficiency would lead to a scram only if there were already a half scram condition. The historical record is that the procedure has been in use for over 10 years, and there has only been 1 event in this time period in which the performance deficiency has led to a scram. Using the conservative assumption that whenever a half-scram precondition existed, the performance deficiency would result in a full scram, the frequency of the preconditions can be estimated based on 1 event in 10 years. The TVA risk analyst used this information to perform a Bayesian IE using a Jeffrey's non-informative prior. This is standard probabilistic risk assessment analytical method. The result of the Bayesian calculation was 0.15/year. This effectively estimated the frequency with the conditions under which the performance deficiency would lead to a reactor scram. This calculation of the IE is conservative, since it ignores the experience from the other Units which use the same procedure and have had no scrams from this type of performance deficiency. If the experience from all three Units was considered then the frequency of the preconditions could have been estimated based on 1 event in 30 years.

The calculated ΔCDF for this performance deficiency is 3.6E-6 times 0.15 or 5.4E-7. Based on this result the performance deficiency meets the threshold to be characterized as "green."

The ΔLERF is calculated by multiplying the change in the IE and the conditional large early release probability (CLERP). The CLERP represents the probabilistic margin to large early release given the occurrence of an initiating event and failures of mitigation equipment.

The CLERP for this performance deficiency was calculated to be 3.7E-7. The change in the IE frequency was calculated to be 0.15/year for reasons stated above. Therefore the ΔLERF for the performance deficiency is 3.7E-7 times 0.15/year or 5.6E-8. Based on this result, the performance deficiency meets the threshold to be characterized as "green."

Summary

This regulatory analysis concludes the following. The failure to follow OI 2-OI- 99 is a performance deficiency. This performance deficiency is more than minor. No identification credit is warranted since the performance deficiency was self revealing. If the performance deficiency is taken in isolation, then it should have screened to “green” without the need to perform a detailed significance determination. However, if plant conditions at the time the performance deficiency occurred are considered, then the standard would be met and the performance deficiency would require a detailed risk analysis. Using ΔCDF and ΔLERF as the figures of merit for the determination of significance, the calculated values for both ΔCDF and ΔLERF meet the criteria to be classified as “green.” The TVA’s calculation of ΔCDF and ΔLERF is conservative because the change in initiating event frequency ignores the experience from the other Units which use the same procedure and have had no scrams from this type of performance deficiency

PRA Evaluation Response:	BFN-2-13-096	Page 1 of 17
NPG Plant(s) and Units(s)	BFN Unit(s): 2	
Department Requesting Evaluation	Licensing – Peter R. Wilson	

Description of Condition to be Evaluated

Perform a significance determination on the following performance deficiency:

On December 22, 2012 failed to correctly follow 2-OI-99 (Operating Instruction) and inadvertently opened the 2A Reactor Protection System (RPS) Motor Generator Breaker. This resulted in the de-energization of the 2A RPS Bus. The loss of power to this bus resulted in a half reactor scram. At the time that this performance deficiency occurred, the 2B RPS bus was also de-energized due an unrelated malfunction that occurred during emergency diesel testing. It was the de-energizing of the 2B RPS bus that led to the operators' use of 2-OI-99 at this time. The loss of power to both RPS busses resulted in a Unit 2 reactor scram and the closure of the Main Steam Isolation Valves (MSIVs). Reactor pressure did not rise to the automatic initiation set point for Safety Relief Valve (SRV) actuation. The Reactor Core Isolation Cooling System (RCIC) and the High Pressure Coolant Injection System (HPCI) reactor water level initiation set point of -45 inches (low low) was reached and the RCIC system and the HPCI system automatically initiated as designed to restore water level above the initiation set point. Both recirculation pumps also tripped on a reactor water level of -45 inches. Reactor pressure control was established by manually operating the SRVs and water level control was established with RCIC system. The HPCI system was returned to standby readiness. The scram was reset, MSIVs were opened, and the Main Condenser was established as a heat sink.

PRA Model Used Including Model Type	Revision 5 of BFN PRA Model (Average T&M Model)		
Initial Unit 2 CDF	N/A	Initial Unit 2 LERF	N/A

Description of Changes made to plant model to evaluate described condition
See attached

Descriptions of the Critical inputs to the evaluations for the assessment to be valid.
N/A

Final Unit 2 ΔCDF	See Attached	Final Unit 2 ΔLERF	See Attached
Additional Metric Used	CCDP/ΔCDF/Color/	Value of Metric	**GREEN**

Risk Management Actions
N/A
Suggested: N/A

Conclusion / Discussion / Description of Application of Risk Insights:
According to the NRC Inspection Manual 0609 since is ΔCDF <1E-6 and ΔLERF is < 1E-7, the risk is considered Green.

Prepared By	Ching Guey / Lance Christiansen Ching Guey/Lance Christiansen /07-18-2013 / 423-751-2480
	Print Name / Signature / Date / Extension
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Enclosure 2

PRA Evaluation Request:

The relevant portion of the draft SRA Analysis produced by the NRC is repeated below in italics for reference purposes.

BACKGROUND (NRC draft analysis)
On December 22, 2012, in response to a loss of Unit 2 RPS bus B related to testing of 3D emergency diesel generator, a Unit Supervisor was dispatched to re-energize the 2B RPS bus. The initial testing that led to the loss of the 2A RPS bus was a test to parallel the 3D and D EDGs. Due to a malfunction in the parallel circuit the two diesels' load sharing function did not work properly and at 11:34 the 3D EDG attempted to reverse power the D EDG. This resulted in loss of the D 4kv shutdown board and the 2B RPS bus. Operators responded to the loss of the 2B RPS bus using procedures AOI-99-1 (Abnormal Operating procedure) and 2-OI-99 (Operating Instruction). 2B RPS bus loss also caused actuation of containment isolation (PCIS) groups 2,3,6, and 8 and a half-trip condition on PCIS group 1. This resulted in a loss of reactor building ventilation which caused a steady increase in main steam vault temperatures. By design, at 189 degrees in the main steam vault, the main steam isolation valves (MSIVs) isolate and a reactor scram signal is generated. Knowledge of this feature imposed time pressure on the operators. The Unit Supervisor had no pre-job brief and no operator peer check person accompanied him to the task. A human error trap existed in procedure 2-OI-99 associated step 5.1 [3] in that it referenced both A and B RPS motor generator set breakers. The listed breakers were only separated by a parenthesis for the B breaker. Thus several factors led to the Unit Supervisor not opening the correct breaker. At time 11:52, eighteen minutes after loss of 2B RPS bus, the Unit Supervisor deenergized the 2A RPS bus causing the unit 2 reactor scram and MSIV closure. This also resulted in a loss of main condenser vacuum and loss of main feedwater.

PERFORMANCE DEFICIENCY
The Unit Supervisor's failure to correctly follow and implement procedure 2-OI-99, Reactor Protection System was a performance deficiency which directly resulted in an automatic reactor scram. This finding was determined to be greater than minor because it was associated with the initiating events cornerstone attribute of human performance, and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during at-power operations.

EXPOSURE TIME
N/A - The performance deficiency caused a trip.

DATE OF OCCURRENCE
December 22, 2012

SAFETY IMPACT
The performance deficiency caused an additional trip. In addition, the secondary heat sink was unavailable during the initial part of the trip response.

RISK ANALYSIS/CONSIDERATIONS

Assumptions
1. *An event evaluation was performed using the ECA module of Saphire 8.*
2. *A transient was assumed, and the MSIVs were modeled closed.*
3. *For the SDP, the increase in risk to the public will be represented by the CCDP of the additional MSIV full closure caused by the performance deficiency.*

4. Resetting the Group 1 isolation, and reopening the MSIVs takes some time, and the steam driven feed pumps would be considered to be not available in the short term for high pressure makeup.
5. In case of failure of the high pressure makeup sources, the fail to depressurize human action, ADS-XHE-XM-MDEPR, OPERATOR FAILS TO DEPRESSURIZE THE REACTOR is currently evaluated at 5.00E-4. The time to core damage is short without makeup. This HRA value is at the lower bound of HRA actions in the model, and can be considered to be a lower bound just due to epistemic uncertainties. Some extra time available to the operators, due to temporary makeup, will not lower the HRA value because of this. Any dependent HRAs will have very limited credit for the same reason.

PRA Model used for basis of the risk analysis: SPAR model draft available for special use for BFNP2 on 4/1/2013. The results were checked for consistency, and the draft model was judged to be better than the model of record.

CALCULATIONS
Results are attached. The CCDP was calculated to be 4.1E-6. The dominant sequences involve loss of the high pressure systems, and a failure to depressurize the vessel by the operators. The dominant cutsets involve fail to run events for HPCI and RCIC. These would delay onset of core damage from about 45 minutes to some longer time, depending on the amount of makeup from the high pressure system. Due to Assumption 4, above, additional HRA credit will not be given for the additional time.

LARGE EARLY RELEASE FREQUENCY IMPACT
Due to scrubbing and plateout in the primary containment, and consideration for the time delay induced by the fail to run sequences, the sequences will be evaluated for LERF with a LERF multiplier of 0.1 for the fail to run sequences. Sequences that do not involve failure of HPI are longer term failures, and evacuation should be effective. For these reasons LERF is considered to be in the mid E-7 range.

***CONCLUSIONS/RECOMMENDATIONS (NRC)** – The risk from the additional trip caused by a full MSIV closure is evaluated to be a White finding for both its CCDP, and LERP.*

PRA Evaluation (TVA):

The evaluation represents the calculation of ΔCDF and ΔLERF given a loss of condenser vacuum. ΔCDF is evaluated by multiplying the change in the initiating event frequency for the loss of condenser vacuum as a result of the performance deficiency by the conditional core damage probability (CCDP) given a loss of condenser vacuum. These two aspects are addressed in the following subsections.

Evaluation of CCDP(CLERP)

The BFN Units 1, 2, and 3 Internal Events PRA peer review of Rev.0 of the BFN PRA model was performed in May, 2009, using the process described in NEI 05-04, Revision 2 (Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard), the ASME and ANS combined PRA Standard, ASME/ANS RA-Sa-2009, and Regulatory Guide 1.200, Revision 2. All findings from the peer review have been dispositioned. Thus, the quality of all elements of the BFN PRA model is sufficient to support "risk significant evaluations with deterministic input".

The BFN PRA CAFTA Model Revision 5 model (an updated version of the peer-reviewed model) was used to evaluate CCDP and CLERP for this evaluation.

To calculate CCDP, the loss of condenser vacuum initiator frequency is set to true; all other initiators are set to false. In accordance with the ground rules for SDP evaluations, the average test and maintenance model was used.

The Revision 5 Internal Events PRA model includes a human action for depressurization of the reactor vessel given an early failure of the high pressure systems (HFA_0001HPRVD1) and for the case where the high pressure systems have run for more than 4 hours (HFA_0001HPRVD1_L). Both of these human actions have the same probabilities (4.5E-4), which is very close to the value used in the SRA's analysis (ADS-XHE-XM-MDEPR, OPERATOR FAILS TO DEPRESSURIZE THE REACTOR is currently evaluated at 5.00E-4). This evaluation also includes a human action to reopen the MSIVs to reestablish the heat sink. A moderate dependence between the operator action to open the MSIVs and depressurization of the reactor vessel via ADS is assumed. This is reasonable given that the two actions have a different purpose. A moderate dependence requires the application of a factor of 0.15 (NUREG-1842, page 3-9) to a combination.

The analysis only credits re-opening the MSIVs when HPCI or RCIC fail late. Therefore, HFA_0001HPRVD1 is kept at its original value of 4.5E-4 while the value of HFA_0001HPRVD1_L is changed to 6.75E-5 (4.5E-5 * 0.15).

<u>Results for CCDP and CLERP:</u>

CCDP was truncated at 1E-12 while CLERP was truncated at 1E-13.

Unit 2 CCDP: 3.6E-6

Unit 2 LERP: 3.7E-7

These results are comparable with the NRC's evaluation using the SPAR model; namely, CCDP is less than 1.0E-5 and greater than 1.0E-6 and the CLERP is less than 1.0E-6 and greater than 1.0E-7.

There appear to be some conservatisms and possible inaccuracies in NRC evaluation of CCDP. For example, all failures, even those in the dominant cut sets that are failures to run of the HPCI and RCIC pumps are assumed to occur at time t=0. This affects the assessment of the likelihood of the operators to perform functions such as initiation of CRD as an injection source, and the initiation of depressurization and biases the evaluation of the corresponding human error probabilities (HEPs) in a conservative direction. It also appears that some cutsets in the dominant sequences contributing to CCDP in the NRC SDP are non-minimal. As one example, for sequence 59, cut set #1 includes the event ADS-XHE-XM-MDEPR with a value of 5E-04, whereas cut set #5 includes the event DEP-XHE-CRD-ADS with an HEP of 1E-04. Given that event tree sequence 59 represents failure of HPI, failure to depressurize and failure of CRD, the latter appears to be correct and the former non-minimal and therefore overestimates the CCDP significantly. Nevertheless, TVA concurs that the CCDP and CLERP are considered to be on the order of 4E-6 and 4E-07 respectively.

Evaluation of ΔCDF and ΔLERF

According to the NRC's Risk Assessment of Operational Events Manual Volume 1 – Internal Events, Revision 2, Initiating Events caused by a performance deficiency are evaluated by converting the CCDPs to ΔCDFs and CLERPs to ΔLERFs by multiplying by a frequency of 1/year. Therefore, the calculated CCDPs and CLERPs are treated as ΔCDFs and ΔLERFs respectively. In other words, the assumption is made that if an initiating event that is in some way associated with a performance deficiency has occurred, the frequency that should be used in the evaluation of ΔCDF or ΔLERF is 1/year.

Based on this method, the December 22, 2012 event would be considered white. As an alternative to the NRC analysis TVA performed an alternate analysis to calculate ΔCDF and ΔLERF, the results of which are presented below.

The two metrics CCDP and ΔCDF serve different purposes. CCDP represents the margin to core damage given the occurrence of an initiating event and is basically a characterization of the ability of the plant to cope with an event. ΔCDF is an assessment of the increase in the likelihood of a core damage event given the existence of a specific condition, or in this case a performance deficiency. ΔCDF can be evaluated using CCDP by multiplying it by the change in initiating event frequency as a result of the performance deficiency. The specific performance deficiency has been determined to have contributed to the initiating event, but it is not the sole cause. In fact, the need to exercise the OI which is the object of the performance deficiency was initiated by an unrelated fault which set up a condition under which the performance deficiency led to a scram. Since the SDP is intended to evaluate the significance under average conditions, the likelihood of this condition needs to be factored in to the evaluation. If this is not done, the evaluation is for the as- found condition at the plant which would be assuming that every time the OI is exercised, a failure would lead to a scram. In other words, the conditioning factors that need to be present for the performance deficiency to result in an initiating event need to be accounted for. ΔCDF can be evaluated using CCDP by multiplying it by the change in initiating event frequency as a result of the performance deficiency.

The following inputs were used:

1. The performance deficiency is characterized by NRC as the failure to properly implement Procedure 2-OI-99, Reactor Protection System. It is noted that that procedure 2-OI-99 is exercised frequently during the course of each year (66 times in the past 5 years according to RCA PER 660862), but only once in the last several years has it led to a reactor trip. This supports the contention that the performance deficiency does not always lead to a reactor trip, but in fact did so under a particular set of conditions. See Attachment 1 for a listing of BFN scrams between the years of 2003 and 2012. Outside of the scram that occurred on December 22, 2012 none of the others were the result of a similar performance deficiency.

2. It is noted that the operating practices have not changed significantly over the years, based on discussions with operators. Given this is the case, then a reasonable and defendable assessment of the impact of the performance deficiency is that it resulted in 1 initiating event in

N years, where N is the number of years during which this procedure has been exercised. Using a Jeffreys non-informative prior, the data of 1 event in N years would results in a mean value of an initiating event frequency of 1.5/N (see Section 6.2.2.5.2 of NUREG/CR-6823). This can be equated to the change in the initiating event frequency as a result of the performance deficiency. This approach effectively estimates the frequency with which the conditions under which the performance deficiency might lead to a reactor trip occur. In this case it is essentially the frequency with which a half scram occurs from an unrelated cause.

3. Based on the discussion with plant operations personnel and review of the 2-OI-99, N is at least 10 years.

The increase in the initiating event frequency due to the performance deficiency is estimated to be 0.15/yr based on at least 10 years use of 2-OI-99 for Browns Ferry Unit 2. Since the same procedures are used in the other units and the same instance hasn't happened there, the initiating event estimated frequency is considered conservative.

Results

The TVA PRA group estimated a CCDP of 3.6E-6. Therefore, TVA estimates ΔCDF due to the deficiency to be approximately 5.4E-7 per year (3.6E-6 x 0.15). ΔLERF is estimated to be approximately 5.6E-8 per year (3.7E-7 x 0.15).

In conclusion, taking into account that the performance deficiency would only lead to a scram under specific conditions (a coincident existing half scram), TVA concludes that both ΔCDF and ΔLERF are below the GREEN threshold of 1.0E-6 and 1.0E-7 per year respectively.

Attachment 1: BFN 1, 2 & 3 Plant Scrams and Initiating Event Category Assignment (from NDN-000-999-2007-0030, Rev. 2, "IE.01 – BFN Probabilistic Risk Assessment – Initiating Event Analysis", Table 17) and 2012 LER Data

Table 1 - BFN 1, 2 & 3 Plant Scrams and Initiating Event Category Assignment					
LER Number	% Power	Event Date	Unit	Category	Event Description
296/2011-003-00	100	28-Sep-11	3	TT	Automatic Reactor Scram due to a Main Turbine Generator Load Reject
296/2011-002-00	0	22-May-11	3	SCRAM	Not included - Reactor Scram Due to Scram Discharge Volume High Water Level
259/2011-001-00	75,75,100	27-Apr-2011	1,2,3	LOOPWE	Three-Unit Scram Caused by Loss of All 500-kV Offsite Power Sources due to Weather
259/2011-005-00	0	27-Apr-2011	1	SCRAM	Not included - Reactor Water Level Scram Due to Distracted Operations Crew
296/2010-004	100	26-Dec-10	3	SCRAM	Manual Reactor Scram Due to High Vibration on the Generator Exciter Inboard and Outboard Journal Bearings
260/2010-003	100	09-Jun-10	2	SCRAM	Reactor SCRAM due to Closure of the Main Steam Isolation valves and Subsequent Invalid RPS SCRAM from the Intermediate Range Monitoring System
260/2009-007	100	29-Sep-09	2	SCRAM	Manual SCRAM during Removal of a Reactor Feedwater Pump from Service
296/2009-001	100	24-Aug-09	3	PLCF	Reactor SCRAM due to Loss of Condensate Booster Pumps
260/2009-004	12	11-Jun-09	2	SCRAM	Technical Specification Shutdown Due to Rise in Unidentified Drywell Leakage
260/2009-006	0	25-May-09	2	SCRAM	Not included - Automatic Reactor Protection System SCRAM While Shutdown
259/2009-001	100	18-Feb-2009	1	TT	Turbine Trip and Reactor Scram Due to Power Load Unbalance Signal on Main Generator
260/2009-001	100	16-Feb-09	2	SCRAM	Manual Reactor Scram Following Stator Cooling Water Equipment Failure
260/2008-01	100	04-Oct-08	2	TT	Automatic Turbine Trip and Reactor Scram Resulting From a Failure of the Design Change Process.
296-2007-05	100	31-Dec-2007	3	TT	Automatic reactor SCRAM due to a generator load reject.
296-2007-03	0	22-Sept-2007	3	SCRAM	Manual reactor SCRAM was followed by the discovery of a leak in an ASME Class I code reactor pressure boundary pipe.
296-2007-01	100	09-Feb-2007	3	PLFW	Automatic reactor SCRAM due to a low reactor water level caused by a loss of feedwater.
260-2007-01	100	11- Jan-2007	2	TT	Automatic turbine trip and reactor SCRAM due to equipment failure during performance of the main generator rheostat test.
296-2006-03	100	29-Aug-2006	3	SCRAM	Manual Scram in Response to Main Turbine Electro-Hydraulic Control (EHC) System Fluid Leak
296-2006-02	100	19-Aug-2006	3	SCRAM	Manual Reactor Scram Due To Loss of the Reactor Recirculation Pumps
296-2005-03	100	31-Oct-2005	3	TT	Reactor Scram from Main Turbine Trip during Switching Operation
296-2005-02	73	17-Sep-2005	3	LCV	Reactor Scram from Main Turbine Trip on Low Condenser Vacuum

Table 1 - BFN 1, 2 & 3 Plant Scrams and Initiating Event Category Assignment					
LER Number	**% Power**	**Event Date**	**Unit**	**Category**	**Event Description**
260/2005-07	100	05-Aug-2005	2	PLFW	Reactor SCRAM due to low reactor water level caused by loss of feedwater pumps.
296-2005-01	100	11-Feb-2005	3	SCRAM	Automatic Reactor Scram Due to False Main Transformer Differential Signal
296-2004-02	100	23-Nov-2004	3	TT	Reactor SCRAM from Main Turbine Trip from loss of all speed feedback
260-2004-02	1	10-Jul-2004	2	SCRAM	Not included, could not occur at power - Automatic reactor SCRAM during startup due to indicated upscale trip on the intermediate range monitors.
260-2004-01	100	8-Jul-2004	2	TT	A main generator load reject condition was spuriously sensed by the main turbine electrohydraulic control (EHC) system.
260-2003-03	63	26-Mar-2003	2	SCRAM	Manual SCRAM of Unit 2 resulting from the 2B reactor recirculation pump trip with OPRM function inoperable.
260-2003-01	0	24-Feb-2003	2	SCRAM	Not included, could not occur at power - Automatic SCRAM resulting from low reactor water level during reactor cooldown.
260-2002-002	100	27-Jul-2002	2	TT	Reactor SCRAM due to Main Bank Transformer Bushing Fault
260-2001-03	100	07/25/2001	2	TT	An automatic SCRAM from 100 percent power due to a main turbine trip from a power-load unbalance that occurred during Combined Intermediate Valve (CIV) testing.
296-2000-05	100	05/24/2000	3	SCRAM	Scram During Level Transmitter Calibration
296-2000-01	70	04/15/2000	3	PLFW	Reactor Scram Due to Feedwater Pump Control Oil System Problem.
260-1999-010	30	09/17/1999	2	TT	Reactor Scram due to Moisture Separator High Level
260-1999-009	100	09/15/1999	2	SCRAM	Manual Reactor Scram due to an EHC leak
260-1999-003	100	05/15/1999	2	TT	Automatic Reactor Scram Due to a Turbine Trip
260-1998-003	100	10/01/1998	2	TT	Reactor Scram from Turbine Trip Due to Failed Isolation Valve in Stator Cooling System
296-1998-003	100	04/07/1998	3	SCRAM	Reactor Manually Scrammed to Prevent Thermal-Hydraulic Instability After Recirculation Pump Runback
260-1997-007	68.9	10/28/1997	2	SCRAM	Reactor Scram Resulting From Pressure Perturbation in the EHC System
260-1997-001	100	04/24/1997	2	TT	Turbine tripped due to a false high reactor water signal as a result of personnel error during surveillance testing.
Additional 2012 BFN Scrams From LER Data					
296/2012-003-01	19	05/22/2012	3	SCRAM	Automatic Reactor SCRAM due to De-Energization of Reactor Protection System from Actuation of 3A Unit Station Service Transformer Differential Relay
296/2012-004-01	1	05/24/2012	3	SCRAM	Manual Reactor SCRAM during Startup due to Multiple Control Rod Insertion
296/2012-005-01	76	05/29/2012	3	SCRAM	Automatic Reactor SCRAM due to an Actuation of a Main Transformer Differential Relay

Table 1 - BFN 1, 2 & 3 Plant Scrams and Initiating Event Category Assignment					
LER Number	**% Power**	**Event Date**	**Unit**	**Category**	**Event Description**
260/2012-006-00	100	12/22/2012	2	SCRAM	Unplanned Automatic Reactor SCRAM due to Loss of Power to the Reactor Protection System

Attachment 2: Unit 2 CCDP-Top 10 Cutsets for Loss of Condenser Vacuum Initiator with Credit for Re-Opening MSIVs

U2_CCDP = 3.636E-06 (Probability)				
#	Cutset Prob.	BE Prob	Inputs	Description
1	3.28E-07	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS1A	
		1.00E+00	FLG_EARLY	EARLY SEQUENCE FLAG
		1.00E+00	FLG_GTRAN_012	SEQUENCE FLAG
		4.50E-04	HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)
		3.20E-02	PRIFD2PMP_0710019	RCIC PUMP FAILS TO START ON DEMAND
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		2.48E-02	TM_2PMP_0730054	HPCI PUMP (UNIT 2) UNAVAILABLE DUE TO TEST AND MAINTENANCE
2	2.68E-07	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS1A	
		1.00E+00	FLG_EARLY	EARLY SEQUENCE FLAG
		1.00E+00	FLG_GTRAN_012	SEQUENCE FLAG
		4.50E-04	HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)
		2.03E-02	PHPFD2PMP_0730054	HPCI PUMP FAILS TO START ON DEMAND

		3.20E-02	PRIFD2PMP_0710019	RCIC PUMP FAILS TO START ON DEMAND
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
3	1.54E-07	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS1A	
		1.00E+00	FLG_EARLY	EARLY SEQUENCE FLAG
		1.00E+00	FLG_GTRAN_012	SEQUENCE FLAG
		4.50E-04	HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)
		1.50E-02	HFL_0071DPIS_CAL	MISCALIBRATION OF RCIC STEAM FLOW INSTRUMENTATION
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		2.48E-02	TM_2PMP_0730054	HPCI PUMP (UNIT 2) UNAVAILABLE DUE TO TEST AND MAINTENANCE
4	1.28E-07	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS1A	
		1.00E+00	FLG_EARLY	EARLY SEQUENCE FLAG
		1.00E+00	FLG_GTRAN_012	SEQUENCE FLAG
		4.50E-04	HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)
		2.03E-02	PHPFD2PMP_0730054	HPCI PUMP FAILS TO START ON DEMAND
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.52E-02	TM_2PMP_0710019	RCIC PUMP (UNIT 2) UNAVAILABLE DUE TO TEST AND MAINTENANCE

5	1.26E-07	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS1A	
		1.00E+00	FLG_EARLY	EARLY SEQUENCE FLAG
		1.00E+00	FLG_GTRAN_012	SEQUENCE FLAG
		4.50E-04	HFA_0001HPRVD1	Failure to initiate reactor-vessel depressurization (transient or ATWS)
		1.50E-02	HFL_0071DPIS_CAL	MISCALIBRATION OF RCIC STEAM FLOW INSTRUMENTATION
		2.03E-02	PHPFD2PMP_0730054	HPCI PUMP FAILS TO START ON DEMAND
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
6	9.23E-08	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS2V	
		1.00E+00	FLG_GTRAN_002	SEQUENCE FLAG
		1.00E+00	FLG_LATE	LATE SEQUENCE FLAG
		6.50E-03	HFA_0023SBCI	Failure to initiate standby coolant injection
		7.40E-04	HFA_0064HWWV	Failure to use hardened wetwell vent for long-term DHR
		1.40E-04	HFA_0074ALIGN_DWS	Failure to align drywell spray and gain spray valve control
		1.20E-05	HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)
		1.00E-05	HFA_0074SPCLATE	Failure to align RHR for suppression pool cooling in the long term
		9.20E-04	HFA_0085ALIGNCST	Failure to align additional inventory for CST - crosstie & levelize CSTs

		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.00E+05	COMBINATION_2452	HEP dependency factor for HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0085ALIGNCST,HFA_0074ALIGN_DWS,HFA_0023SBCI,HFA_0064HWWV
		1.00E+05	COMBINATION_2452A	HEP dependency factor for COMBINATION_2452
		1.00E+05	COMBINATION_2452AA	HEP dependency factor for COMBINATION_2452A
		1.35E+00	COMBINATION_2452AAA	HEP dependency factor for COMBINATION_2452AA
7	9.21E-08	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS2A	
		1.00E+00	FLG_GTRAN_001	SEQUENCE FLAG
		1.00E+00	FLG_LATE	LATE SEQUENCE FLAG
		6.20E-04	HFA_0064DWVENT	OPERATOR FAILS TO INITIATE DRYWELL VENT
		7.40E-04	HFA_0064HWWV	Failure to use hardened wetwell vent for long-term DHR
		1.40E-04	HFA_0074ALIGN_DWS	Failure to align drywell spray and gain spray valve control
		1.20E-05	HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)
		1.00E-05	HFA_0074SPCLATE	Failure to align RHR for suppression pool cooling in the long term
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.00E+05	COMBINATION_2946	HEP dependency factor for HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0074ALIGN_DWS,HFA_0064DWVENT,HFA_0064HWWV
		1.00E+05	COMBINATION_2946A	HEP dependency factor for COMBINATION_2946
		1.30E+03	COMBINATION_2946AA	HEP dependency factor for COMBINATION_2946A

8	9.21E-08	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS2V	
		1.00E+00	FLG_GTRAN_002	SEQUENCE FLAG
		1.00E+00	FLG_HWWV_SUCCESS	HARDENED WETWELL VENT SUCCESS
		1.00E+00	FLG_LATE	LATE SEQUENCE FLAG
		6.50E-03	HFA_0023SBCI	Failure to initiate standby coolant injection
		1.40E-04	HFA_0074ALIGN_DWS	Failure to align drywell spray and gain spray valve control
		1.20E-05	HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)
		3.80E-03	HFA_0074RHR_CST	OPERATOR FAILS TO ALIGN RHR PUMPS TO CST
		9.20E-02	HFA_0074SDC_ALIGN	OPERATORS FAILS TO ALIGN SDC
		1.00E-05	HFA_0074SPCLATE	Failure to align RHR for suppression pool cooling in the long term
		3.50E-04	HFA_0075CSCST	OPERATOR FAILS TO ALIGN CORE SPRAY PUMPS TO CST
		8.50E-04	HFA_0085MAXCRD	Failure to maximize CRD flow for RPV injection
		4.50E-04	HFA_0LPIINIT30	Failure to establish low-pressure injection given loss of high pressure injection
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.00E+05	COMBINATION_2547	HEP dependency factor for HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0LPIINIT30,HFA_0074ALIGN_DWS,HFA_0085MAXCRD,HFA_0023SBCI ,HFA_0074RHR_CST,HFA_0075CSCST,HFA_0074SDC_ALIGN
		1.00E+05	COMBINATION_2547A	HEP dependency factor for COMBINATION_2547
		1.00E+05	COMBINATION_2547AA	HEP dependency factor for COMBINATION_2547A
		1.00E+05	COMBINATION_2547AAA	HEP dependency factor for COMBINATION_2547AA

		1.96E+02	COMBINATION_2547AAAA	HEP dependency factor for COMBINATION_2547AAA
9	9.21E-08	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS2V	
		1.00E+00	FLG_GTRAN_002	SEQUENCE FLAG
		1.00E+00	FLG_LATE	LATE SEQUENCE FLAG
		6.50E-03	HFA_0023SBCI	Failure to initiate standby coolant injection
		7.40E-04	HFA_0064HWWV	Failure to use hardened wetwell vent for long-term DHR
		1.40E-04	HFA_0074ALIGN_DWS	Failure to align drywell spray and gain spray valve control
		1.20E-05	HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)
		1.00E-05	HFA_0074SPCLATE	Failure to align RHR for suppression pool cooling in the long term
		8.50E-04	HFA_0085MAXCRD	Failure to maximize CRD flow for RPV injection
		4.50E-04	HFA_0LPIINIT30	Failure to establish low-pressure injection given loss of high pressure injection
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.00E+05	COMBINATION_2462	HEP dependency factor for HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0LPIINIT30,HFA_0074ALIGN_DWS,HFA_0085MAXCRD,HFA_0023SBCI,HFA_0064HWWV
		1.00E+05	COMBINATION_2462A	HEP dependency factor for COMBINATION_2462
		1.00E+05	COMBINATION_2462AA	HEP dependency factor for COMBINATION_2462A
		3.24E+03	COMBINATION_2462AAA	HEP dependency factor for COMBINATION_2462AA

10	9.19E-08	1	%2LCV	LOSS OF CONDENSER VACUUM
		9.20E-01	CRIT_FACTOR	
		1.00E+00	FLG_CLASS2V	
		1.00E+00	FLG_GTRAN_002	SEQUENCE FLAG
		1.00E+00	FLG_LATE	LATE SEQUENCE FLAG
		1.30E-03	HFA_0002RPV_LVL	OPERATOR FAILS TO MAINTAIN RPV LEVEL
		6.50E-03	HFA_0023SBCI	Failure to initiate standby coolant injection
		7.40E-04	HFA_0064HWWV	Failure to use hardened wetwell vent for long-term DHR
		1.40E-04	HFA_0074ALIGN_DWS	Failure to align drywell spray and gain spray valve control
		1.20E-05	HFA_0074HPSPC1	Failure to align RHR for suppression pool cooling (non-ATWS/IORV)
		1.00E-05	HFA_0074SPCLATE	Failure to align RHR for suppression pool cooling in the long term
		8.50E-04	HFA_0085MAXCRD	Failure to maximize CRD flow for RPV injection
		9.99E-01	SRVFC2PCV_0SORV	NO SRVS STICK OPEN
		1.00E+05	COMBINATION_1738	HEP dependency factor for HFA_0002RPV_LVL,HFA_0074HPSPC1,HFA_0074SPCLATE,HFA_0074ALIGN_DWS,HFA_0085MAXCRD,HFA_0023SBCI,HFA_0064HWWV
		1.00E+05	COMBINATION_1738A	HEP dependency factor for COMBINATION_1738
		1.00E+05	COMBINATION_1738AA	HEP dependency factor for COMBINATION_1738A
		1.12E+03	COMBINATION_1738AAA	HEP dependency factor for COMBINATION_1738AA

PRA Evaluation Request	
NPG Plant(s) and Unit(s)	BFN Unit 2
Department Requesting Evaluation	CNL

Description of Condition to be Evaluated

Perform a significance determination following performance deficiency:On December 22, 2012 failed to follow 2-OI-99 (Operating Instruction) and inadvertently opened the 2A Reactor Protection System (RPS) Motor Generator Breaker. This resulted in the de-energization of the 2A RPS Bus. The loss of power to this bus resulted in a half reactor scram (need to get other impacts of the loss of bus). At the time that this performance deficiency occurred, the 2B RPS bus was also de-energized due an unrelated malfunction that occurred during emergency diesel testing. The loss of power to both RPS busses resulted in a Unit 2 reactor scram and the closure of the Main Steam Isolation Valves (MSIVs). Reactor pressure did not rise to the automatic initiation set point for Safety Relief Valve (SRV) actuation. The Reactor Core Isolation

Comprehensive List of Specific Equipment Impacted

Include if component is Unable to perform its design basis function

Primary Heat Sink

Date/Time Evaluation Needed	7/15/2013	
Requested By	Peter Wilson/7/11/13	
	Print Name / Signature / Date / Extension	Supervisor's Initials
Approved By (Corporate PRA Manager)	Ching Guey / Ching Guey / 7/17/2013	
	Print Name / Signature / Date	

TVA 41117 Page 1 of 1 NPG-SPP-09.11-2 [01-07-2011]

Enclosure 2