May 21, 2003

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FROM: Mark F. Reinhart, Chief ***/RA/***
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SUBJECT: RESULTS OF THE BRAIDWOOD NUCLEAR POWER STATION UNITS 1 AND 2 SIGNIFICANCE DETERMINATION PROCESS PHASE 2 NOTEBOOK BENCHMARKING VISIT

During January, 2003 NRC staff and contractors visited the Exelon Generation Company in Warrenville, IL to compare the Braidwood Nuclear Power Station Significance Determination Process (SDP) Phase 2 notebook to the licensee's risk model results to ensure that the SDP notebook was generally conservative. The Braidwood Probabilistic Safety Assessment (PSA) did not include external initiating events, therefore no sensitivity studies were performed to assess the impact of these initiators on SDP color determinations. In addition, the results from analyses using the NRC's draft Revision 3i Standardized Plant Analysis Risk (SPAR) model for Braidwood were also compared with the licensee's risk model. The results of the SPAR model benchmarking effort will be documented in next revision of the SPAR (revision 3) model documentation.

The benchmarking visit identified that there was generally good correlation between the Phase 2 SDP Notebook and the licensee's PSA. The results indicate that the Braidwood SDP Phase 2 notebook was more conservative in comparison to the licensee's PSA. A summary of the results of comparisons of hypothetical inspection findings between SDP notebook and the licensee's PSA are as follows.

12% Underestimates Risk Significance
50% Matches Risk Significance
29% Overestimates Risk Significance by 1 Order of Magnitude
2% Overestimates Risk Significance by 2 Orders of Magnitude
7% Unable to compare with licensee's PSA.

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The Rev. 1 SDP notebook was improved as a result of the benchmarking activity. The percentage of cases that the Rev. 1 SDP notebook matched that of the updated licensee's PSA increased from 38% to 50% and the percentage of overestimates was reduced from 36% to 31%. The percentage of underestimates remained the same at 12%.

During the benchmarking, some characteristics of the Braidwood PSA were noted which contributed to the difference in results between the SDP notebook and the plant PSA. These characteristics include the following:

- Dependency among the operator actions are evaluated and the associated error probability is defined based on previous actions. As such, the dependent Human Error Probabilities (HEPs) in the Braidwood PSA are very high compared to the dependent HEPs in other plants of similar design. The licensee plans to reevaluate these dependent human actions.

- The medium and large loss-of-coolant accident (LOCA) frequencies are approximately one order of magnitude lower compared to the industry averages. The steam generator tube rupture (SGTR) frequency is high considering the steam generators had been replaced.

- The Main Steam Line Break (MSLB) initiator does not address the pressurized thermal shock considerations. This is typical of most of the PSAs.

- In SGTR scenarios, the plant PSA does not consider the need for refueling water storage tank (RWST) makeup where secondary cooling has failed, but high pressure recirculation is successful following successful bleed and feed. The licensee plans to reevaluate and analyze this scenario.

The licensee's PSA staff was very knowledgeable of the plant model and provided very helpful comments during the benchmark visit.

The attachment describes the process and results of the comparison of the Braidwood SDP Phase 2 Notebook and the licensee's PSA.

Attachments: As stated

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Accession Number: ML031260699

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SUMMARY REPORT ON BENCHMARKING TRIP

TO THE BRAIDWOOD NUCLEAR POWER STATION

UNITS 1 AND 2

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April 2003

ATTACHMENT

Table of Contents

List of Tables

1. INTRODUCTION

A Benchmarking of the Risk-Informed Inspection Notebook for the Braidwood Nuclear Power Station, Units 1 and 2, to be referred to as Braidwood, was conducted during a plant site visit on January 13-15, 2003. The Nuclear Regulatory Commission (NRC) staff (R. Gibbs, M. Parker, R. Perch, and P. Wilson) and Brookhaven National Laboratory (BNL) staff (P. Samanta) participated in this Benchmarking exercise.

In preparation for the meeting, BNL staff reviewed the SDP notebook for the Braidwood Nuclear Power Station and evaluated a set of hypothetical inspection findings using the Rev. 0 SDP worksheets. In addition, NRC staff provided the licensee with a copy of the meeting protocol.

The major milestones achieved during this meeting were as follows:

1. Recent modifications made to the Braidwood PRA were discussed for consideration in the Rev. 1 model to be prepared following benchmarking.

2. Importance measures, including the Risk Achievement Worths (RAWs) for the basic events in the internal events model for average maintenance, were obtained from the licensee.

3. Benchmarking was conducted using the Rev. 0 SDP model and the revised SDP model considering the licensee's input and other modifications that were judged necessary based on comparison of the SDP model and the licensee's detailed model.

4. For cases where the color evaluated by the SDP notebook differed from that determined based on the RAW values generated by the updated licensee's PRA, results of the licensee's base case model including the dominant minimal cutsets were reviewed to understand the reason for the differences.

The Rev. 1 version of the SDP notebook was developed considering the changes identified based on the licensee's input and the evaluation of the benchmarking results.

2. SUMMARY RESULTS FROM BENCHMARKING

Summary of Benchmarking Results

Benchmarking of the SDP Notebook for the Braidwood Nuclear Power Station, Units 1 and 2 was conducted comparing the risk significance of the inspection findings obtained using the notebook with that to be obtained using the plant PRA. The benchmarking identified the hypothetical inspection findings for which the results of the evaluation using the notebook were under or overestimations compared to the plant PRA. Five cases of non-conservative results or underestimation by one color by the notebook (i.e., the significance obtained by the notebook is one color lower than that to be obtained by the plant PRA) were noted. Also, one case of a conservative result by two orders of magnitude (i.e., the significance obtained using the notebook is two colors higher than that to be obtained using the plant PRA) was noted. A summary of the results of the risk characterization of hypothetical inspection findings is as follows:

12% (5 of 42 cases)	Non-conservative; underestimation of risk significance (by one order of magnitude)
2% (1 of 42 cases)	Conservative; overestimation of risk significance (by two orders of magnitude)
29% (12 of 42 cases)	Conservative; overestimation of risk significance (by one order of magnitude)
50% (21 of 42 cases)	Consistent risk significance
7% (3 of 42 cases)	Results could not be compared.

Detailed results of Benchmarking are summarized in Table 1. Table 1 consists of eight columns. The first two columns identify the components or the case runs. The assigned colors from the SDP Rev. 0 worksheets without incorporating any modification from the Benchmarking exercise are shown in the third column. The fourth column gives the basic event name in the plant PRA used to obtain the risk achievement worth (RAW) for the component out of service or the failed operator action. The fifth and sixth columns respectively show the licensee's internal RAW value and the color to be defined based on the RAW values, from the latest PRA model. The seventh column presents the colors for the inspection findings based on the Rev. 1 version of the notebook. The Rev. 1 version of the notebook is prepared considering the revisions to the Rev. 0 version of the SDP notebook judged applicable during Benchmarking. The last column provides comments identifying the difference in results between the SDP Rev. 1 notebook and the plant PRA, and the applicable rules in obtaining the color of the inspection finding using the SDP notebook.

Table 2 presents a summary of the comparisons between the results obtained using the Braidwood Nuclear Power Station notebook and the plant PRA. It also shows a comparison of the results using the Rev. 0 and Rev. 1 versions of the notebook. The results show that overestimations by the notebook were reduced and the matches were increased through revisions to the notebook implemented as a result of Benchmarking. The overestimations were reduced from 36% to 31% and the matches increased from 38% to 50%. The percentage of underestimations (12%) remained unchanged. However, over and underestimations by more than an order of magnitude also were reduced.

Observations on the Braidwood PRA

During the benchmarking, some characteristics of the Braidwood PRA are noted which contribute to the difference in results between the SDP notebook and the plant PRA.

1. In the Braidwood PRA, dependency among the operator actions are evaluated and the associated error probability is defined based on previous actions. For example, failure to conduct bleed and feed is assigned a human error probability (HEP) of 0.5 given failure to start alternate feedwater. Otherwise, failure to conduct bleed and feed has a HEP of approximately 6.5E-3. Also, following failure to depressurize RCS, the HEP for failure to switch over to high pressure recirculation (HPR) is 0.1. The dependent HEPs in the Braidwood PRA, are considered very high compared to the dependent HEPs in other plants of similar design. The licensee plans to reevaluate these dependent human actions.

2. The medium and large LOCA frequencies are approximately an order of magnitude lower compared to the industry averages. The SGTR frequency is high considering that the Steam Generators were replaced.

3. The Main Steam Line Break (MSLB) initiator does not address the pressurized thermal shock considerations. This is typical of most of the PRAs.

4. In SGTR scenarios, the plant PRA does not consider the need for RWST makeup where secondary cooling has failed, but high pressure recirculation is successful following successful bleed and feed. The licensee plans to reevaluate and analyze this scenario.

Discussion of Non-conservative Results by the Notebook

During benchmarking, non-conservative results or underestimations by the notebook compared to the plant PRA were noted for 5 out of the 42 cases analyzed. In all 5 cases, the underestimation was by one order of magnitude, i.e., by one color. The specific items are: 1 MDAFW pump, 1 battery, 1 Auxiliary spray valve, operator failure to depressurize late in a SGTR event, and operator failure to restart MFW pump/PCS. These cases were further analyzed comparing the detailed minimal cutsets from the case by case computer runs using the plant PRA. The reasons for the difference, as identified, are discussed below for each of the items.

1. Failure of the MDAFW pump was underestimated by one color by the notebook. Given a failure of the MDAFW pump, secondary heat removal by auxiliary feedwater is accomplished in this plant by the diesel-driven AFW pump. In the notebook, the diesel-driven AFW pump is given a credit of 1 train which is equivalent to 1E-2. In the plant PRA, the diesel-driven AFW pump has a combined unavailability of approximately 4E-2. This difference in diesel-driven AFW unavailability in multiple sequences has contributed to the underestimation.

2. Failure of a battery was underestimated by one color by the notebook. In this plant, since the battery charger can carry the SI loads given a failure of the battery, the failure of the battery plays a significant role in LOOP scenarios where the battery charger is not available. In these scenarios, when the loss of the battery leads to a loss of the MDAFW pump, the remaining capability is due

to the diesel-driven AFW pump. As explained above, there is approximately a factor of four difference in unavailability of the diesel-driven AFW pump between the notebook and the plant PRA which contributes to the underestimation. In addition, the LOOP frequency is approximately a factor of two higher in the plant PRA compared to the value used in evaluation using the notebook. It is placed in Row II in the notebook representing a value of 1E-2 per reactor-year.

3. Failure of an auxiliary spray valve was underestimated by one color by the notebook. Auxiliary spray is used for late depressurization in a SGTR event. The SGTR frequency in this plant is approximately a factor of 7 higher compared to the generic value used in evaluations using the notebook. This difference contributed to the underestimation.

4. Operator failure to depressurize late in a SGTR event was underestimated by one color by the notebook. The reason for this underestimation is essentially the same as that for the auxiliary spray valve discussed in item 3 above. The difference in SGTR initiating frequency contributed to the underestimation.

5. Operator failure to restart the MFW pumps/PCS was underestimated by one color by the notebook. The reason for this difference is the human error probability (HEP) used in the plant PRA for failure to bleed and feed in transients following failure to restart the MFW pumps. The HEP used in the PRA is 0.5. Considering this high HEP value, an operator action credit of 1 is used in the notebook for feed and bleed following failure of MFW/PCS. Still, the difference in the HEP value results in the underestimation. The licensee plans to review this HEP along with other dependencies among the human actions modeled in the PRA.

Discussion of Conservative Results by the Notebook

Twelve cases of overestimations or conservative results were noted during the benchmarking. Of the twelve, one case was overestimated by two orders of magnitude and the remaining eleven cases were overestimated by one order of magnitude. Since the notebooks are designed to be screening tools and include assumptions that can result in conservative assessment, overestimation by an order of magnitude, i.e., by one color, is not unexpected. We focus on the items that are conservative by two orders of magnitude. We discuss these cases first and then provide general discussions for the conservative results by the notebook.

1. Failure of 1 SG PORV was overestimated by two orders of magnitude, i.e., by two colors. SG PORVs provide steam relief for the secondary and are used for RCS depressurization. In all cases when SG PORVs are used, multiple redundancies are available. Usually, 1/4 SG PORVs is needed or 1/5 steam SVs for 2 of 4 SGs are needed. In the PRA calculation, the loss of 1 SG PORV has a minimal impact. However, in the notebook evaluation, many sequences are counted considering the base case impact leading to the overestimation by two orders of magnitude.

The major differences that have generally contributed to the overestimations by the notebook can be summarized as follows:

1. Medium and Large LOCA frequencies in the plant PRA are lower compared to the generic values used in the SDP notebook. The frequencies used in the plant PRA are 2.76E-5 and 3.31E-6 per reactor-year respectively for medium and large LOCAs. In the notebook, they are respectively placed in Rows IV and V. In the evaluations using the notebook, the approximate respective frequencies are 1E-4 and 1E-5 per reactor-year.

2. Following loss of battery charger, batteries are credited to carry the load for some duration. In the SDP notebook, such details are not modeled and batteries are assumed to fail. This leads to some conservatism in some cases, particularly in case of battery chargers.

3. In ATWS scenarios, the success criteria in the plant PRA for primary pressure relief changes based on the core lifetime. In the notebook, the conservative success criteria is used leading to conservative results for the failure of the SRVs.

<u>Changes Incorporated Following Benchmarking Resulting in Updating of Benchmarking Results</u>

Following benchmarking, operator action credit for conducting emergency boration in an ATWS was changed to 1. This was based on the licensee's HEP of 8.62E-2 for the action. This change did not result in any change in the colors assessed for the inspection findings in the benchmarking.

Table 1. Summary of Benchmarking Results for Braidwood Nuclear Power Station, Units 1 and 2

Internal Events CDF = 3.3E-05, excluding internal flooding, at Truncation Level of 1E-10
RAW Thresholds are: W = 1.03 (RAW), Y = 1.30 (RAW), R = 4.03 (RAW)

No.	Component Out of Service or Failed Operator Action	SDP Before	Basic Event Name	RAW	Plant CDF Color	SDP After	Comments
	Component						
1.	1 MDAFW pump	R	1AF01PA-----PMFR	20.49	R	Y	Under
2.	1 DDAFW pump	R	1AF01PB-----PDFR	7.01	R	R	
3.	1 CCW pump	W	1CC01PA-----PMFR	2.07	Y	Y	RAW does not include impact on initiating event frequency.
4.	1 Cond. pump	G	Truncated	~1.00	G	G	
5.	1 MD Feed pump	G	1FW01PA-----PMMM	1.01	G	G	
6.	1 SG PORV	Y	1MS0018A----PVCC	1.00	G	Y	Over (double)
7.	EDG IA	Y	1DG1A-------DGFR	1.11	W	Y	Over. EDG 1A supports the MDAFW pump.
8.	EDG1B	Y	1DG1B-------DGFR	1.19	W	W	
9.	4 KV ESF Bus	R	1AP141------BSLP	118.13	R	R	
10.	1 DC Bus	R	1DC111AF2---DBCO	86.39	R	R	
11.	1 battery	Y	1DC111------BYLP	5.71	R	Y	Under. At Braidwood, battery charger can carry the SI loads. Inspection finding on the battery is assessed using LOOP, LEAC, and LDC111 or LDC112 worksheets.
12.	1 battery charger	Y	1DC112------BCLP	1.5	Y	R	over
13.	1 SI pump	W	1SI01PA-----PMFR	1.01	G	W	over
14.	1 Charging pump	Y	1CV01PB-----PMMM	1.05	W	Y	over
15.	1 Accumulator	Y	1SI8956A----CVCC	1.10	W	Y	over

No.	Component Out of Service or Failed Operator Action	SDP Before	Basic Event Name	RAW	Plant CDF Color	SDP After	Comments
16.	1 RHR pump	R	1RH01PB-----PMFS	2.58	Y	Y	
17.	1 RHR HX	Y	Truncated; separate CDF run	1.1	W	Y	over
18.	1 ESW pump	R	1SX01PB-----PMFR	1.35	Y	Y	
19.	1 IA compressor	Y	1SA01C------CMFR	1.00	G	G	
20.	1 MSIV	Y	Truncated; separate CDF run	1.03	W	R	The RAW from plant PRA is considered not comparable. The plant PRA does not model pressurized thermal shock (PTS) concerns.
21.	1 Non-Essential SW pump	G	0WS01PA-----PMFR	1.30	Y	Y	
22.	1 PORV	Y	1RY456------PVCC	1.73	Y	Y	
23.	1 SRV fto	W	1RY8010A----ZVCC	1.01	G	W	over
24.	Stuck-open PORV	Y	1RY455A-----PVOO	1.33	Y	Y	
25.	1 Block Valve	W	1RY8000A----BVOO 1RY8000B----BVCC	1.26	W	W	
26.	1 RCFC unit	W	Truncated; separate CDF run	1.00	G	W	over
27.	1 Safe Shutdown Sequencer Train	W	Truncated; separate CDF run	1.01	G	W	over
28.	1 BAT Pump	___	Truncated; separate CDF run	1.01	G	G	
29.	1 Aux. Spray valve	___	Truncated; separate CDF run	1.09	W	G	under
30.	1 Fire Pump	Y	0FP03PA----PMFS	1.07	W	W	

No.	Component Out of Service or Failed Operator Action	SDP Before	Basic Event Name	RAW	Plant CDF Color	SDP After	Comments
	Operator Actions						
31.	Fails FB	R	1RY-FB-GTR-HPVCA 1RY-FB-GTR-HPVOA 1RY-FB-SGTRHPVCA 1RY-FB-SLOCHPVCA 1SI-FB-AF—HSYCA 1SI-FB-----HSYOA	12.45	R	R	
32.	Fails to initiate HPR	R	1SI-HPR----HSYOA 1SI-HPR-ML-HSYCA 1SI-HPR-NR-HSYCA	13.85	R	R	
33.	RCS Cooldown in SLOCA	Y	1RC-DS-SLOCHDVOA	1.85	Y	R	over
34.	REC2	G	NA	NA	NA	G	Braidwood calculates a probability of core Uncovery Before power Recovery (UBR). These UBRs are a function of the power recovery curves and time to core damage given the boundary conditions of the sequence (e.g., AF success or failure). Comparable RAW from PRA is not available.
35.	REC5	W	NA	NA	NA	W	See above
36.	Sec. Dep. In SGTR	Y	1RC-DP-SGTRHSYOA	3.83	Y	Y	
37.	Late Depressurization in SGTR	___	Separate CDF run	5.27	R	Y	under

No.	Component Out of Service or Failed Operator Action	SDP Before	Basic Event Name	RAW	Plant CDF Color	SDP After	Comments
38.	Emergency Boration	W	1RC-EB-ATWSHSYCA 1RC-EB-ATWSHSYOA	1.05	W	W	
39.	RCP trip and switch from VCT to RWST	Y	1CV-ALL----HPMOA	6.27	R	R	
40.	ESW Cross-tie to other unit	R	0SX-XTIE---HMVOA	1.15	W	Y	over
41.	RCP trip and Cross-tie fire water	R	1FP-PRI-7X-HMVOA	2.55	Y	R	over
42.	Fails to restart PCS	Y	1FW-ALT-GTRHSYOA 1FW-ALTSGTRSYOA	42.0	R	Y	under

Table 2: Comparative Summary of Benchmarking Results

Comparisons	Rev. 0 SDP Notebook		Following Benchmarking	
	Total Number of Cases Compared = 42			
	Number of Cases	Percentage	Number of Cases	Percentage
SDP: Less Conservative	5 (1)	12	5	12
SDP: More Conservative	15 (2)	36	13 (3)	31
SDP: Matched	16	38	21	50
Comparable RAW not available or not modeled in the Notebook	6 (4)	14	3 (5)	7

Notes:

1. One case by two orders of magnitude and the remaining 4 cases by one order of magnitude.

2. Three cases by two orders of magnitude and the remaining 12 cases by one order of magnitude.

3. One case by two orders of magnitude and the remaining 12 cases by one order of magnitude.

4. Three cases could not be evaluated by the Rev. 0 version of the notebook; they were not modeled. Comparable RAWs were not available for the remaining 3 cases.

5. Comparable RAWs were not available for these three cases.

3. PROPOSED MODIFICATIONS TO THE REV. 0 SDP NOTEBOOK

A set of modifications were proposed for the Rev. 0 SDP notebook as a result of the site visit. These proposed modifications are driven by the licensee's revisions to the plant's PRA, better understanding of the current plant design features, revised Human Error Probabilities (HEPs), modified initiator frequencies, and the results of benchmarking.

3.1 Specific Changes to the Rev. 0 SDP Notebook for the Braidwood Nuclear Power Station, Units 1 and 2

The following changes were made based on the licensee's inputs and evaluations conducted as part of Benchmarking:

1. Changes to Table 1

    1.1 Loss of Instrument Air (LIA) is added in Row III.

    1.2 Loss of a DC Bus (LDC) is replaced by two separate initiators: Loss of DC Bus 111 (LDC111) and Loss of DC Bus 112 (LDC112) in Row IV.

    1.3 Loss of Component Cooling Water (LCCW) was moved from Row V to Row IV and a footnote was added justifying its placement in Row IV.

2. Changes to Table 2

    2.1 The following systems/components are added in the table: AMSAC, BAT pumps, Auxiliary Spray, and Fire pumps.

    2.2 ESFAS dependency is added for motor-driven and diesel-driven AFW pumps, SI pumps, and RHR pumps.

    2.3 120 VAC and SX dependencies are added for diesel-driven AFW pump. SX is the main cooling source and alternate water source.

    2.4 125 VDC dependency is removed for 480 VAC power system.

    2.5 125 VDC dependency is added for 120 VAC instrument power system.

    2.6 Safe shutdown sequencer is added as a support system for Charging pumps.

    2.7 SX is included as a support system for RHR pumps. SX supports RHR pump lube oil cooling and cubicle coolers.

2.8 It is noted that essential service water (SX) pumps can be cross-tied to the other unit. 480 VAC dependency of the SX pumps is noted.

2.9 It is noted that failure of both DC buses is needed for failure of MSIVs.

2.10 Additional changes are made for clarification and footnotes are modified as applicable.

3. Changes to Worksheets and Event Trees

3.1 Credit for bleed and feed (BF) was changed to operator action = 1 when it followed failure to restart MFW. The plant PRA uses a HEP of 0.5, but the licensee plans to reevaluate it. This high HEP was considered unrealistic and an operator action credit of 1 instead of 0 was used in all applicable cases.

3.2 The credit for HPR was changed from operator action = 2 to operator action = 3 based on the plant-specific HEP.

3.3 SLOCA event tree and worksheet were modified to need BF following failure of SGPORV. In addition, MFW was renamed PCS and SGPORV was redefined as RCSDEP, similar to Westinghouse 4 loop plants.

3.4 In the SLOCA worksheet, HPR was redefined as two functions, HPR1 and HPR2. HPR1 is assigned a credit of operator action = 1 and applies following failure to RCSDEP. HPR2 applies for other situations and is assigned a credit of 3.

3.5 SORV event tree and worksheet are modified similar to SLOCA.

3.6 MLOCA event tree and worksheet are modified to remove need for AFW, MFW, SGPORV, and LPR.

3.7 LLOCA event tree and worksheet are modified to combine RCFC and LPR functions similar to the LPR function in SLOCA. Need for RCFCs are included within the LPR function.

3.8 SGTR event tree was modified and a revised worksheet was incorporated. Along with pressure equalization using secondary system, late depressurization was modeled. Also, RWST makeup was modeled.

3.9 In the SGTR function, the mitigation capability for EQ was defined to require 2/3 SG PORVs or 2/3 SG steam dump valves.

3.10 In the ATWS worksheet, Turbine Trip (TTP) function was defined as 1 multi-train system. AMSAC is a multi-train system.

3.11 In the ATWS worksheet, the mitigation capability for emergency boration was revised to include RWST or 1/2 BAT pumps.

3.12 MSLB event tree and worksheet were modified to include pressurized thermal shock concerns when more than 1 MSIV fails to close. Revised event tree and worksheet consistent with other plants of similar design are included.

3.13 In the LCCW worksheet, for the SWP function, 1/2 charging pumps is included in the mitigation capability description.

3.14 In the LSW worksheet, the XTFP function mitigation capability was revised to include 1/2 fire pumps and the operator action credit was changed from 1 to 2.

3.15 Separate worksheets were developed for loss of DC Bus 111 (LDC111) and loss of DC Bus 112 (LDC112).

3.16 LEAC event tree and worksheet were modified considering the changes made to the SORV event tree and worksheet.

3.17 Loss of instrument air (LIA) worksheet was added.

3.2 Generic Change in 0609 for Inspectors

Based on the lessons from this benchmarking, recommendations for improving 0609 are as follows:

1. Diesel-driven pumps, similar to steam-driven pumps, tend to have an unavailability greater than 1E-2. In this plant, the diesel-driven AFW pump has an unavailability of approximately 4E-2. In SDP notebooks, these pumps are treated similar to other hardware trains and are assigned a credit of 1 train. This difference contributes to some underestimations. Assigning a credit of 1, similar to steam-driven trains, can be considered to avoid underestimations.

2. The SGTR initiating event is currently assigned to Row III in Table 1 of the notebook for Pressurized Water Reactors (PWRs). For many plants, the plant-specific frequency is at the high end of Row III which leads to some underestimations. Considerations may be given to placing SGTR in Row II for older plants where Steam Generators have not been replaced. For the Byron and Braidwood plants, the plant PRAs need to revise the SGTR frequency reflecting the replaced Steam Generators. Keeping SGTR in Row III is justified because Steam Generators have been replaced.

3.3 Generic Change to the SDP Notebook

None identified.

4. DISCUSSION ON EXTERNAL EVENTS

An integrated external event PRA model was not available for the Braidwood Nuclear Power Station. No evaluation was conducted for the external event risk during the Benchmarking exercise.

5. LIST OF PARTICIPANTS

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