August 27, 2003

MEMORANDUM TO: Stuart Richards, Chief
Inspection Program Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Patrick D. O'Reilly
Operating Experience Risk Applications Branch
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

FROM: Mark F. Reinhart, Section Chief **/*RA*/**
Licensing Section
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

SUBJECT: RESULTS OF THE MONTICELLO NUCLEAR GENERATING PLANT PHASE 2 NOTEBOOK BENCHMARKING VISIT

During June, 2003, NRC staff and contractors visited the Monticello Nuclear Generating Plant to compare the LaSalle Significance Determination Process (SDP) Phase 2 notebook and licensee's risk model results to ensure that the SDP notebook was generally conservative. The Monticello PRA did not include most external initiating events (only fire initiators); so 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 Standard Plant Analysis Risk (SPAR) model for Monticello were compared with the licensee's risk model. The results of the SPAR model benchmarking effort will be documented in the next revision of the SPAR (revision 3) model documentation.

The benchmarking visit identified that there was not a strong correlation between the Phase 2 SDP Notebook and the licensee's PRA. The results indicate that the Monticello Phase 2 notebook was more conservative in comparison to the licensee's PRA. The revision 1 SDP notebook will capture 65% (results matched or overestimated the licensee's PRA by one order of magnitude) of the risk significance of inspection findings. A summary of the results of comparisons of hypothetical inspection findings between the SDP notebook and the licensee's PRA are as follows.

Attachments: As stated

CONTACT: Peter Wilson, SPSB/DSSA/NRR
301-415-1114

8% Underestimates Risk Significance
35% Match Risk Significance
30% Overestimates Risk Significance by 1 Order of Magnitude
19% Overestimates Risk Significance by 2 Orders of Magnitude
8% Overestimates Risk Significance by 3 Orders of Magnitude

The lack of a strong correlation between the notebook and the Monticello PRA was due to significant differences in PRA modeling and assumptions. The major differences between the plant PRA and the SDP notebook are summarized below.

1. Early CRD injection

   The Monticello PRA credited control rod drive system (CRD) as an early injection source similar to HPCS and RCIC. Based on the calculations conducted by the plant, sufficient flow could be achieved by the CRD pumps to avoid core damage without the success of any other high pressure injection sources (high pressure coolant injection and reactor core isolation cooling systems). As a rule, in the BWR SDP notebooks CRD is never credited as an early injection source, but it is considered adequate as a late injection following success of other injection sources..

2. Late injection

   The Monticello PRA credited late injection following containment failure to avert core damage. The licensee conducted an evaluation to determine whether the conditions in the reactor building following containment failure would be severe enough to disable equipment. At the Monticello plant, with the exception of certain valves and piping, equipment for the RHR/LPCI and Core Spray (CS) systems are located in the RHR corner rooms. The licensee stated that the RHR and CS pumps could operate in 100% humidity and at pressures far exceeding any that the reactor building can sustain. Based on these considerations, late injection following containment failure was credited in the Monticello PRA. SDP notebook models of this BWR typed, do not credit such late injection following containment failure.

3. Vessel rupture due to SRVs failing to open

   The Monticello PRA modeled an Excessive LOCA (vessel rupture) due to all 8 SRVs failing to open. The probability for all 8 SRVS failing to open was 1.19E-04. This resulted in dominant cutsets for many initiators where the likelihood of such an excessive LOCA is feasible. The SDP notebooks do not model such scenarios since the initiating frequency typically used by the industry and NRC is about three orders of magnitude smaller.

4. CS as success in vessel rupture

   Following a vessel rupture due to SRVs failing to open, Monticello considered that core damage could be averted with the success of the core spray system. In the SDP notebooks, vessel rupture is assumed to lead to core damage. This difference in

assumption resulted in core spray pumps having higher risk significance in the Monticello PRA.

5. HRA dependencies

The Monticello PRA considered dependencies among different human actions using a logic diagram. In this approach, complete dependencies were assumed among many actions; i.e., failure in one action may result in failure of other actions with a probability of 1. For example, failure to align suppression pool cooling would result in operator failing to depressurize for late injection. This was different from the way credits for operator actions are considered in the SDP notebooks. This difference resulted in differences in risk significance between the Monticello PRA and the SDP notebook.

The Rev-1 SDP notebook has been slightly improved as a result of the benchmarking activity. The number of cases that the Rev-1 SDP would match that of the updated licensee's PRA has increased from 8 to 13. The number of over estimations dropped from 24 to 21 cases. In addition, the number of underestimations decreased from 5 to 3.

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

Attachment A describes the process and results of the comparison of the Monticello SDP Phase 2 Notebook and the licensee's PRA.

Attachments: As stated

5. HRA dependencies

The Monticello PRA considered dependencies among different human actions using a logic diagram. In this approach, complete dependencies were assumed among many actions; i.e., failure in one action may result in failure of other actions with a probability of 1. For example, failure to align suppression pool cooling would result in operator failing to depressurize for late injection. This was different from the way credits for operator actions are considered in the SDP notebooks. This difference resulted in differences in risk significance between the Monticello PRA and the SDP notebook.

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NAME	*PWilson:nxh2	*SBurgess	MReinhart
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SUMMARY REPORT ON BENCHMARKING TRIP TO

MONTICELLO NUCLEAR GENERATING PLANT

(June 3-5, 2003)

P.K. Samanta
Brookhaven National Laboratory
Upton, NY 11973

July 2003

ATTACHMENT A

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1. INTRODUCTION

A Benchmarking of the Risk-Informed Inspection Notebook for the Monticello Nuclear Generating Plant was conducted during a plant site visit on June 3 to 5, 2003. NRC staff (O. Hopkins, M. Parker, and P. Wilson) and BNL staff (P. Samanta) participated in this Benchmarking exercise.

In preparation for the meeting, BNL staff reviewed the SDP notebook for the Monticello Nuclear Generating Plant 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 Monticello Nuclear Generating Plant 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 event 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 based on the RAW values generated by the updated licensee's PRA, a judgment about the difference was made based on the detailed base case results available for the plant. Minimal cutsets evaluating the impact of the hypothetical inspection findings were reviewed to identify the reasons for the differences.

The Rev. 1 version of the Monticello Nulcear Generating Plant SDP notebook was prepared based on the lessons learned from the benchmarking at the site. Benchmarking identified some differences in modeling assumptions between the Monticello plant PRA and the SDP notebook. These differences contribute to the differences in risk significance to be obtained for inspection findings when the results of the SDP notebook and the plant PRA are compared. The differences identified during benchmarking are discussed in this report.

2. SUMMARY RESULTS FROM BENCHMARKING

Summary of Benchmarking Results

Benchmarking of the SDP notebook for the Monticello Nuclear Generating Plant (MNGP) was conducted by comparing the order-of-magnitude results obtained using the notebook with that obtained using the plant-specific PRA. Cases for which the SDP notebook results were under or overestimates were identified. Three cases of a conservative result by three orders of magnitude (i.e., the significance obtained using the notebook is three colors higher than that to be obtained using the plant PRA), seven cases of conservative results by two orders of magnitude, and eleven cases of conservative results by one order of magnitude were noted. In addition, three cases of underestimation (one by two orders of magnitude and two by one order of magnitude) were noted. A summary of the results of the risk characterization of hypothetical inspection findings is as follows:

8% (3 of 37 cases)	underestimation of risk significance
8% (3 of 37 cases)	overestimation of risk significance by three orders of magnitude
19% (7 of 37 cases)	overestimation of risk significance by two orders of magnitude
30% (11 of 37 cases)	overestimation by one order of magnitude
35% (13 of 37 cases)	consistent risk significance.

Detailed results of Benchmarking are summarized in Table 1. Table 1 consists of eight columns. The first two columns identify the components/failed operator actions or the case runs. The assigned colors from the Rev. 0 SDP 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 out of service component 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 following Benchmarking. The last column provides comments identifying the difference in results between the Rev. 1 SDP notebook and the plant PRA.

Table 2 presents a summary of the comparisons between the results obtained using the Monticello Nuclear Generating Plant notebook and the plant PRA. It also shows a comparison of the results using the Rev. 0 and Rev. 1 versions of the notebook. It can be noted that there are differences in the assessed risk significance of inspection findings between the SDP notebook and the plant PRA, as revealed by the benchmarking. In approximately 35% of the cases, the assessed risk significance by the SDP notebook was non-conservative or conservative by more than one order of magnitude. In the remaining 65% of the cases, the assessed risk significance matched or was within an order of magnitude. The reasons for the differences were analyzed and are discussed below. Rev. 0 version of the notebook resulted in differences in even higher percentage of cases. In only 51% of the cases the assessed risk significance would have been comparable or within one order of magnitude.

Monticello PRA Modeling and Assumptions

The differences in Monticello PRA modeling and assumptions, and the SDP notebook modeling and assumptions contributed to differences in the risk significance noted in the benchmarking. In this section, the major differences between the plant PRA and the SDP notebook are summarized.

1. Early CRD injection

The Monticello PRA credited CRD as an early injection source similar to HPCS and RCIC. Based on the calculations conducted by the plant, sufficient flow could be achieved by the CRD pumps to avoid core damage without the success of any other high pressure injection sources (HPCI or RCIC). In the SDP notebook, CRD was not credited as an early injection source, but was considered adequate as a late injection following success of other injection sources. This difference in the modeling assumption had important implications for the risk significance of the inspection findings. Because of this assumption, the risk significance of HPCI, RCIC, depressurization function, loss of condenser, etc, were lower and risk significance of the CRD pump was higher in the licensee’s PRA. Following the USNRC’s evaluation of the licensee’s calculation for justifying crediting CRD as an early injection source, SDP notebooks may be adjusted with similar credits.

2. Late injection

The Monticello PRA credited late injection following containment failure to avert core damage. The licensee conducted an evaluation to determine whether the conditions in the reactor building following containment failure would be severe enough to disable the equipment there. At the Monticello plant, with the exception of certain valves and piping, equipment for the RHR/LPCI and Core Spray (CS) systems are located in the RHR corner rooms. Based on the licensee’s evaluation, the condition in this room would remain mild enough for these pumps to operate. The licensee stated that the RHR and CS pumps could operate in 100% humidity and at pressures far exceeding any that the reactor building can sustain. Based on these considerations, late injection following containment failure was credited in the Monticello PRA. The SDP models for this type of BWR plant does not credit such late injection following containment failure. This difference in assumption also resulted in differences in risk significance difference between the plant PRA and the SDP notebook. RHR and RHRSW pumps, RHR HX, Containment Venting function, and containment heat removal (CHR) function were shown to be of higher risk significance in the notebook because of this difference.

3. Vessel rupture due to SRVs failing to open

The Monticello PRA modeled an Excessive LOCA due to all 8 SRVs failing to open. The probability for all 8 SRVS failing to open was 1.19E-04. This resulted in dominant cutsets for many initiators where the likelihood of such an excessive LOCA is feasible. The SDP notebooks do not consider such scenarios.

4. CS as success in vessel rupture

Following a vessel rupture due to SRVs failing to open, Monticello considered that core damage could be averted with success of core spray system. In the SDP notebooks, vessel rupture is

assumed to lead to core damage. This difference in assumption resulted in core spray pumps having higher risk significance in the Monticello PRA.

5. HRA dependencies

Monticello PRA considered dependencies among different human actions using a logic diagram. In this approach, complete dependencies were assumed among many actions, i.e., failure in one action may result in failure of other actions with a probability of 1. For example, failure to align suppression pool cooling would result in operator failing to depressurize for late injection. This was different from the way credits for operator actions are considered in the SDP notebooks. This also resulted in differences in risk significance between the Monticello PRA and the SDP notebook.

<u>Discussion of Non-conservative Results or Underestimations by the Notebook</u>

Three cases of underestimation or non-conservative results were noted during the Benchmarking. They related to the CRD pump, CS pump, and the RBCCW pump. The reasons for these underestimations are discussed below:

1. Inspection finding on a CRD pump was underestimated by two orders of magnitude by the notebook compared to the plant PRA. The reason for this underestimation was due to the difference in assumption between the plant PRA and the notebook. The assumption related to crediting CRD as an early high pressure injection source, as discussed above. Crediting CRD as an early injection made its role more important in the plant PRA.

2. Inspection finding for a CS pump was underestimated by one color using the notebook compared to the plant PRA. The reason here also was due to the difference in assumption between the plant PRA and the notebook. The modeling of vessel rupture following failure of SRVs to open and successful operation of CS pumps as a means to avert core damage in such a scenario made the role of CS pump more significant in the plant PRA.

3. Inspection finding for 1 RBCCW pump was also underestimated by one color using the notebook compared to the plant PRA. RBCCW pump cools the CRD pumps, and the reason for its underestimation was the same as that for the CRD pump as discussed above.

<u>Discussion of Conservative Results by the Notebook</u>

Twenty one cases of overestimation (three cases by three colors, seven cases by two colors, and eleven cases by one color) were noted during Benchmarking. The reasons for these overestimations primarily related to the differences in assumptions between the notebook and the plant PRA, as discussed earlier.

An overestimation by three colors was noted for the PCS steam component, CV vent path, and operator failing to vent. The reasons for these three orders of magnitude overestimations are discussed below.

1. An overestimation by three colors was noted for PCS steam component. In a loss of PCS scenario, the dominant risk contributors were the loss of high pressure injection systems and the operator's failure to depressurize. As discussed earlier, the Monticello PRA credited CRD as a high pressure injection source which was not done in the SDP notebook. In addition, the human error probability (HEP) for depressurization in the Monticello PRA is 1.6E-4, which is an order of magnitude lower than the credit of 3 used in the notebook.

2. An overestimation by three orders of magnitude was noted for the CV path. The reason for this overestimation was due to the difference in crediting late injection following containment failure. Monticello PRA's assumption that following containment failure, late injection could be conducted using the RHR and CS pumps made the significance of containment venting lower. In addition, the failure to initiate suppression cooling had a lower likelihood in the PRA compared to the credits in the notebook.

3. Similar to the containment vent path, operator failing to vent the containment was overestimated by three orders of magnitude. The reason for this difference was exactly the same as that for the CV path discussed above.

Overestimation by two orders of magnitude was noted for seven cases: 1 SRV failing to open, 1 RHR pump, 1 RHRSW pump, 1 RHR HX, Valve MO 2014 failing to open, 1 battery charger, and operator failing to initiate SPC. The reasons for these overestimations are discussed below.

1. 1 SRV failing to open was overestimated by two orders of magnitude by the notebook. The plant PRA credited late injection following containment failure and credited continued use of LPCI and CS pumps following controlled venting. These made the significance of a stuck-open SRV lower.

2. Inspection finding for 1 RHR pump was overestimated by two orders of magnitude. The dominant contributors to the risk significance of the RHR pump related to the reduced capability of suppression pool cooling (SPC). These core damage sequences are associated with containment venting and failure. The credit for continued use of LPCI and CS pumps following controlled venting and for late injection following containment failure resulted in the overestimation. Also, there are four RHR pumps with one pump needed for success. The credit for the four pumps in the notebook was "1 multi-train system", but the unavailability for the loss of the four pumps was lower.

3. Inspection finding on 1 RHRSW pump was also overestimated by two orders of magnitude. A RHRSW pump was needed for cooling the RHR heat exchanger during SPC. The configuration and success criteria of RHRSW pumps were the same as that of the RHR pumps. The reason for the overestimation was similar to that discussed for the RHR pumps.

4. Inspection for 1 RHR HX, similar to the RHR and RHRSW pumps, was overestimated by two orders of magnitude. Loss of 1 RHR HX reduced the SPC mitigation capability to a single train system. The differences in assumption for the

remaining mitigation capability, the same as that discussed for the RHR and RHRSW pumps, contributed to the overestimation.

5. Valve MO 2014 is the injection valve in the LPCI path. Failure to open this valve was overestimated by two orders of magnitude. In the plant PRA, RHR and CS pumps were assumed to be available following successful venting. In the notebook, however, alternate injection sources were needed. For the loss of the injection valve, crediting of the RHR and CS pumps following containment venting resulted in the difference in the risk significance.

6. The battery charger was overestimated by two orders of magnitude. In the plant PRA, the loss of a battery charger would be promptly detected and the alternate charger aligned to the affected bus. The HEP for failing to align the alternate charger was low. In the SDP notebook evaluation, a recovery credit of 1 was assigned resulting in the overestimation.

7. Failure to initiate SPC was also observed to be overestimated by two orders of magnitude. The reason for this overestimation was similar to that for other items affecting containment heat removal; namely, RHR and RHRSW pumps. Again, the difference in assumption for late injection following containment failure resulted in the overestimation.

The reasons for the overestimation by one color in eleven cases were also due to the differences in assumptions discussed earlier. No additional reasons were identified. These items are not separately discussed.

Changes Incorporated Following Benchmarking Resulting in Updating of Benchmarking Results

Following benchmarking, the following additional change was made to the notebook:

In the LSW worksheet, credit for CSW pump train was removed since the loss of service water would result in loss of instrument air leading to failure of the AOV in the CSW system. In this single train system, this failure would result in failure to inject using the CSW pump. This change did not change the colors obtained during benchmarking.

Table 1: Summary of Benchmarking Results for Monticello Nuclear Generating Plant

Internal Events CDF is 6.47E-6 per reactor-year excluding internal flooding at a 1E-11 truncation limit
RAW thresholds(1) are W =1.15, Y = 2.55, R = 16.45, RR=155.6

No.	Component Out of Service or Failed Operator Action	SDP Worksheet Results (Before)	Monticello Nuclear Plant Basic Event	Monticello Nuclear Plant RAW Ratio	Color by Monticello Nuclear Plant RAW	SDP Worksheet Results (After)	Comments
	Component						
1.	HPCI	R	HPTP209XXS12	2.01	W	Y	over by 1order of magnitude
2.	RCIC	Y	IPTP207XXS12	1.91	W	Y	over by 1 order of magnitude
3.	PCS steam	R	GVRRV4923L	1.05	G	R	over by 3 orders of magnitude
4.	PCS feed: 1 MD FW pump	G	FPFP2AXXS12	1.28	W	Y	over by 1 order of magnitude
5.	1 Condensate pump	W	FPAP1AXXXR12	1.0	G	W	over by 1 1order of magnitude
6.	1 SRV fto	R	XVRSSRVCCN88	1.0	G	Y	over by 2 orders of magnitude
7.	1 SRV ftc	Y	XVRONESRVC	1.26	W	Y	over by 1 order of magnitude

No.	Component Out of Service or Failed Operator Action	SDP Worksheet Results (Before)	Monticello Nuclear Plant Basic Event	Monticello Nuclear Plant RAW Ratio	Color by Monticello Nuclear Plant RAW	SDP Worksheet Results (After)	Comments
8.	1 CS pump	G	CPCP208AXR12	1.94	W	G	under by 1 order of magnitude
9.	1 RHR pump	Y	RPRP202AXR14	1.07	G	Y	over by 2 orders of magnitude
10.	1 RHR HX	R	RHXE200AXF	1.46	W	R	over by 2 orders of magnitude
11.	1 RHRSW pump	Y	SPWP109AXR14	1.0	G	Y	over by 2 orders of magnitude
12.	Valve MO 2014 FTO	R	RVMM02015N12	1.03	G	Y	over by 2 orders of magnitude
13.	1 SW pump	Y	SPSP102AXR13	1.12	G	W	over by 1 order of magnitude
14.	1 SLC pump	G	LPLP203AXS12	1	G	G	
15.	RPT 1 train trip device	G	LCBFB11AXN12	1	G	G	
16.	RPT both trains (trip device)	Y	LCB11ABCCN22	1.34	W	Y	over by 1 order of magnitude
17.	1 IA compressor	R	NCMCMP14CM	1.14	G	W	over by 1 order of magnitude
18.	1 EDG	Y	ADGEDG11XR13	3.6	Y	Y	

No.	Component Out of Service or Failed Operator Action	SDP Worksheet Results (Before)	Monticello Nuclear Plant Basic Event	Monticello Nuclear Plant RAW Ratio	Color by Monticello Nuclear Plant RAW	SDP Worksheet Results (After)	Comments
19.	Onsite AC power- DG#13	G	ADLDG13XXR13	1.46	W	W	
20.	1 EDG-ESW pump	Y	SPEP111AXS12	5.68	Y	Y	
21.	1 EAC Bus (Bus 15)	RR	ABS15XXXXG	58.37	R	R	
22.	DC Panel D-111	RR	DBSD111XXG	35.98	R	R	
23.	DC Battery 11	RR	DBAD100XXR12	58.8	R	R	
24.	1 Battery Charger	Y	DBCD13XXXR	1.05	G	Y	over by 2 orders of magnitude
25.	1 CRD pump	W	JPJP201AXR12	2.97	Y	G	under by 2 orders of magnitude
26.	1 RBCCW pump	W	BPBP6AXXXR12	1.17	W	G	under by 1 order of magnitude
27.	1 CV path	R	VENT-CHR-Y	1.0	G	R	over by 3 orders of magnitude
28.	1 DD Fire pump	W	YPDP105XXR	1.37	W	W	
29.	1 MD Fire pump	W	YPYP104XXS12	1.0	G	G	

No.	Component Out of Service or Failed Operator Action	SDP Worksheet Results (Before)	Monticello Nuclear Plant Basic Event	Monticello Nuclear Plant RAW Ratio	Color by Monticello Nuclear Plant RAW	SDP Worksheet Results (After)	Comments
30.	1 SP vacuum breaker	G	ZVB238AXC18	1.0	G	G	
	Operator Actions						
31.	Fails PCS	Y	FW-CNTRL-Y	1.8	W	Y	over by 1 order of magnitude
32.	Fails to DEP	RR	DEP-HOUR-Y	133.65	R	RR	over by 1 order of magnitude
33.	Fails to initiate SPC	RRR	RHR-DHR-Y	91.26	R	RRR	over by 2 orders of magnitude
34.	Fails to CV	R	VENT-CHR-Y	1.0	G	R	over by 3 orders of magnitude
35.	Fails to use fire pumps	W	CASE RUN	1.43	W	W	
36.	Fails to operate DG-13	G	DG13-BFD-Y	1.2	W	W	
37.	Fails ro recover Offsite power (6 hrs.)	W		NA(2)		W	Not evaluated
38.	Fails to initiate SLC	Y	SLC-INI-Y	1.46	W	Y	over by 1 order of magnitude
39.	Fails to INH	Y		NA(2)		Y	Not evaluated

Note:

1. RAW threshold are obtained considering delta-CDF impacts of 1E-6 for White (W), 1E-5 for Yellow (Y), 1E-4 for Red (R), 1E-3 for Double Red (RR), and 1E-2 for Triple Red (RRR). RR implies that the CDF impact of the finding is between 1E-3 and 1E-2, and RRR implies that the CDF impact is between 1E-2 and 1E-1.

2. NA means that RAW for the item could not be obtained from the Licensee's PRA.

Table 2: Comparative Summary of the Benchmarking Results

		SDP Worksheets (Rev. 0)		SDP Worksheets Modified (Rev. 1)	
		Number of Cases	**Percentage**	**Number of Cases**	**Percentage**
SDP: Non-Conservative		5	14	3	8
SDP: Conservative by	1 order	11	29	11	30
	2 orders	7	19	7	19
	3 orders	6	16	3	8
SDP: Matched		8	22	13	35
Total			100	37	100
Not evaluated or comparable RAW not available		2		2	

3. ADDITIONAL PROPOSED MODIFICATIONS TO SDP WORKSHEETS

3.1 Specific Changes to the Rev. 0 SDP Worksheets for Monticello Nuclear Generating Plant

The changes made to the Monticello Nuclear Generating Plant notebook to develop the Rev. 1 version during and after the plant onsite benchmarking visit are summarized here and are also included in the updated notebook.

<u>Changes made to Monticello Nuclear Generating Plant Rev. 0 Notebook to complete the Rev. 1 Notebook</u>

1. Changes to Table 1

   1.1 Loss of AC Bus 15 (LAC15) and Loss AC Bus 16 (LAC16) initiators were added to Row III.

   1.2 A footnote was added stating that the LLOCA initiating frequency is 1.6E-4 per reactor-year, higher than the generic industry-wide frequency used in the notebooks.

2. Changes to Table 2

   2.1 Footnotes were added to clarify the 125 VDC, 250 VDC, and Instrument N2 dependencies and the available backups for SRVs.

   2.2 Power Conversion system (both steam and feedwater side) and their support systems were defined.

   2.3 Instrument AC dependency to HPCI and RCIC was clarified. CST was removed as a support system to these systems.

   2.4 IA dependency to CRDH was deleted.

   2.5 A footnote was added stating that on loss of 125 VDC RHR, CS, and RHRSW pumps can be manually operated.

   2.6 Major components for containment venting were revised to include hard pipe vent only. Instrument N2 was noted as a support system for hard pipe vent.

   2.7 Separate rows were defined for the EDGs and EDG-HVAC. EDG-HVAC was noted as a support system for EDGs.

   2.8 A footnote was added defining the number of battery chargers in the DC system.

   2.9 It was noted that three SW pumps are respectively supported by Div 1, Div 2, and non-safety AC buses. Two of the pumps are supported by the Div 2 125 VDC.

2.10 Emergency Service Water (ESW) system was deleted from the Table. The system supports room cooling and acts as backup to the SW system. Its risk significance is always Green.

2.11 It is noted that one of the air compressors has its own cooling, with DG#13 as the power source.

2.12 Room Cooling was removed from the table. None of the pump room cooling is required for the 24 hour period.

3. Changes to the Worksheets and Event trees

3.1 LPCI mode of the RHR pumps when credited was defined as 1 train because of the use of the loop selection logic at the plant in all applicable worksheets. Based on the pressure differential between the loops, the loop selection logic will always select a train. However, all 4 pumps are credited.

3.2 Late Inventory (LI) was revised to denote that use of CRD does not require depressurization in all applicable worksheets. Credits were modified to include an operator action credit of 1 for the CRD pumps, and an operator action credit of 2 for the remaining low pressure injection sources.

3.3 The mitigation capability of the CHR function was redefined to better represent the number of pumps and the train configurations.

3.4 In the TRANS worksheet, LPI mitigation capability was revised to include the condensate service water (CSW) pumps. Also, condensate pumps were deleted from LI.

3.5 In the TPCS worksheet, LPI mitigation capability was revised to include the CSW pumps. Condensate pumps were added to the LI function.

3.6 In the SLOCA worksheet, LPI mitigation capability was revised to include the CSW pumps. Condensate and CRD pumps were deleted from the LI function. Operator action credit for using PCS was revised to 2.

3.7 SORV worksheet and event tree were redefined noting that the reactor will depressurize after some time and LPI will be needed. CSW pumps were added as a LPI source. CRD was credited, but condensate pumps were removed, from the LI function.

3.8 MLOCA worksheet and event tree were modified to delete containment venting. Injection sources are assumed to fail prior to containment venting. Use of feed and condensate pumps are deleted. Operator action credit for depressurization was changed to 2 because of the need to depressurize within 15 mins.

3.9 Similar to MLOCA, the LLOCA worksheet and event trees were modified to delete containment venting. Credit for condensate pumps for LI was deleted.

3.10 In the LOOP worksheet, credit for recovery of offsite power within 6 hrs was revised from 2 to 1. The operator action credit for using the fire pump as an alternate LPI was changed to 2 (from 0). LI mitigation capability was revised to state 1/1 diesel fire pump.

3.11 In the ATWS worksheet, mitigation capability for OVERP function was revised from 7/8 SRVs to 6/8 SRVs. Operator action credit for INH was changed from 1 to 2, and the mitigation credit for RPT was changed to 1 multi-train system considering the redundancy of the trip breakers.

3.12 LSW worksheet was modified to include the capability to vent and use LI. Hard pipe vent is not affected by the loss of SW.

3.13 LOIA worksheet was also revised to credit capability to vent and use LI. CRD pumps are credited in LI since they are not affected by IA.

3.14 LDCA worksheet was modified to include CSW as a LPI source. LI was modified to include injection using the fire pumps. Loop A injection can obtain power from the other bus.

3.15 LDCB worksheet was modified to include CSW as a LPI injection source.

3.16 Two worksheets were added for Loss of AC Bus 15 (LAC15) and loss of AC Bus 16 (LAC16). For LAC16, one of the feed pumps is lost in addition to loss of 1 train of safety systems.

3.2 Generic Changes in IMC 0609 for Guidance to NRC Inspectors

None.

3.3 Generic Change to the SDP Notebooks

Based on lessons of benchmarking at Monticello, two items were noted for consideration for generic changes to the notebooks for BWR plants:

1. As noted here, the Monticello PRA credits CRD as an early high pressure injection source. SDP notebooks currently do not credit CRD as an early high pressure injection source unless plants are specifically designed with an enhanced CRD, e.g, BWR/6 plants. Some other plants are also taking credits for CRD as an early high pressure injection. Based on review of the evaluations conducted by these plants, guidelines for crediting CRD as an high pressure injection may need to be defined for SDP notebooks. This difference between the plant PRA and SDP notebook assumption can impact the risk significance of a number of components and operator actions.

2. Monticello PRA also credits continued use of LPCI and CS pumps following controlled venting. Continued use of these pumps following controlled venting, including use of the non-hardened vent path, can be considered on a plant-specific basis.

4. DISCUSSION ON EXTERNAL EVENTS

Monticello Nuclear Generating Plant does not have an integrated external event PRA. The licensee noted that their IPEEE study for fire events was quantitative, but the IPEEE seismic analysis used the non-quantitative seismic margins method. Other external events were also qualitative.

ATTACHMENT 1. LIST OF PARTICIPANTS

Ogabana Hopkins	USNRC/NRR
Mike Parker	USNRC/Region III
Peter Wilson	USNRC/NRR
Pranab Samanta	BNL
Robert Buell	INEEL
Timothy Wellumsen	NSP