Attachment C

CRDR 2715709 SWMS Attached Media for Unit 2 Trip

Note: Whenever possible an ADOBE .pdf should be created from electronic files directly. If only hardcopy is available, then documents can be scanned for these records.

The Investigation Director or designee shall arrange for the following **Standard Evaluations** and include in SWMS Attached Media for the CRDR

	Item	*Attachment*	*Name*
✔	1	C	Plant Transient Review Assessment
✔	2	C	Safety Limit Review Evaluation
✔	3	C	Plant Performance Evaluation
✔	4	C	Plant Protection System Response Evaluation
✔	5	C	Control System Response Evaluation
✔	6	C	Nuclear Safety Assessment

Note that items 1, 3, and 4 are included in the STA letter.

Digitally signed by: McDowell, James P(Z98774)
Date: 06/22/2004 14:16:34
Reason: I have reviewed this document and verified included signatures.
Location: PVNGS

C/15

APS

Company Correspondence

DATE: June 17, 2004

Memo# 294-01900-DWV
TO: Jim McDowell
Sta. # 7997
Ext. # 5668

FROM: Don Vogt
Sta. # 7833
Ext. # 5926

Digitally signed by: Vogt, Donald W(Z96416)
Date: 06/18/2004 16:13:47
Reason: I am the author of this document
Location: PVNGS

SUBJECT: Unit 2 Loss of Off-Site Power, Main Generator Trip, Reactor Trip, DG-A voltage regulation failure of 6/14/2004

The Shift Technical Advisors have conducted a Plant Performance of the Unit 2 Loss of Offsite Power Reactor Trip and "Alert" Emergency Classification due to loss of Diesel Generator 'A' on 06/14/2004.

The Plant Performance Evaluation is an overall evaluation of how the plant responded to the Loss of Offsite Power Reactor Trip. The STAs perform this evaluation of the plant's response within two distinct areas. The Safety Function Impact is an analysis of the transient's impact on each of the PVNGS Safety Functions. The General Plant Performance is an assessment and description of equipment malfunctions, abnormal alarms and/or events observed during the course of the event.

The evaluations are performed utilizing information from ERFDADS plots, control board strip chart recorders, alarm-typer outputs (PMS/RONAN), operating procedures, and personnel statements. The summary of these evaluations is provided as attachments.

Attachments
Event Summary/General Plant Response
Plant Performance Evaluation/ Plant Transient Response

cc:	D. Smith	7602	D. Carnes	7997	T. Radtke	7294
	D. Mauldin	7605	C. Seaman	7636	M. Shea	7299
	M. Winsor	7669	F. Riedel	7894	P. Borchert	7904
	P. Kirker	7398	M. Grigsby	7298	M. McGhee	7198
	P. Wiley	7848	J. Hesser	7002	J. T. Taylor	7848

Email: Shift Technical Advisors (7833)

Unit 2 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

EVENT SUMMARY

Unit 2 was operating steady state at 100% power (based on MFW flow venturi's) and had no evolutions in progress. At 07:41, switchyard problems resulted in a loss of offsite power to NAN-S03 and NAN-S04 and a Main Turbine Trip. The 13.8 KV buses (NAN-S01 and NAN-S02) received a load shed command; all four RCPs stopped and the reactor tripped on Low DNBR. Both DGs received a start signal and powered up their respective 4.16 KV bus. DG 'A' closed in on PBA-S03, and after approximately 20 seconds voltage was lost on PBA-S03 and DG 'A'. Although red light indication was observed for the PBA-S03 loads, the loss of non-class power resulted in a loss of flow indication for the 'A' train SP, EW, etc. With no SP cooling flow, DG 'A' was emergency shutdown and this combination (LOOP + loss of DG 'A') resulted in the Alert Emergency Classification at 07:54. Post-trip, the Unit 2 operating crew initiated a MSIS as directed by the LOOP/LOFC EOP (at 07:51:58) and a plant cooldown via ADVs to control Pressurizer Level <56%. After restoration of offsite power at 09:51, the Emergency Plan was exited. Evaluations will be documented in CRDR 2715709.

Initial Sequence of Events

Time	Event - LOOP; Low DNBR Reactor Trip; Loss of DG 'A'– Alert EAL
07:40:56	Main Generator Negative Sequence Alarm (MAYS14)
07:41:07	Main Transformer 'B' Trouble Alarm ((MAYS51)
07:41:08	Main Transformer 'A' Trouble Alarm (MAYS31)
07:41:08	CEDM MG Set 'B' Input Breaker OPEN (SFYS30)
07:41:11	PBA-S03 Trouble Alarm (NBYS18)
07:41:13	PBB-S04 Trouble Alarm (NBYS21)
07:41:15	LOP/Load Shed PBBS04 (SAYS10) and DG 'B' Start Signal (SAYS12) LOP/Load Shed PBAS03 (SAYS 9) and DG 'A' Start Signal (SAYS11)
07:41:17	CEDM MG Set 'A' Input Breaker OPEN (SFYS29)
07:41:17	13.8 KV Switchgear 1 and 2 Load Shed complete
07:41:17	Unit 2 Main Generator 525KV Breaker 935 OPEN (MAZS9)
07:41:17	RCP 1A and RCP 2A Not Running (RCYS13 and RCYS15)
07:41:17	Przr Relief Valve Not Closed (RCYS21) and Przr Relief Valve Monitor Power Fail (RCYS20). Main Steam Relief Valve Position Open (SGYS12) and Main Steam Relief Valves Monitor Power Fail (SGYS11).
07:41:17	Main Turbine CIV 1-6 Not Open (MTZS580/590/600/910/920/930)
07:41:17	RPS Channels A-D (SBYS21-24)
07:41:17.141	Fast Closing of Intercept Valves Commanded
07:41:17.385	Low DNBR Channel 'C' Trip (SBTC04)
07:41:17.407	Low DNBR Channel 'B' Trip (SBTB04)
07:41:17.471	Reactor Trip Circuit Breaker 'B' OPEN (SBMTB10)
07:41:17.472	Reactor Trip Circuit Breaker 'C' OPEN (SBMTC10)
07:41:17.493	The 4th CEDM Power Bus Undervoltage Alarm (SFCE9UV1-4)
07:41:17.523	No ETSV Pressure Trip (MTYS47)
07:41:17.526	Mechanical Overspeed Trip (MTYS16)
07:41:17.581	Master Turbine Trip (MTYS21)
07:41:17.667	Generator/Reactor Initiated Trip (MTYS27)
07:41:18	RCP 1B and RCP 2B Not Running (RCYS14 and RCYS16)
07:41:18	Pressurizer PSV200 & 201 Relief Temperature High (RCTS107 & 108)
07:43:11	Unit 2 Main Generator 525KV Breaker 938 OPEN (MAZS8)
07:43:50	Diesel Generator 'A' and 'B' Status is RUNNING (DGYS5 and DGYS6).
07:43:52	PBBS04 Bus Voltage is normal (PBYS11)
07:43:56	ESF Bus Undervoltage Channel 'B' Not Tripped (SAYS23-26)

Unit 2 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

07:43:59	BOP-ESFAS Load Sequencer 'A' in "LOP" Mode (SAYS41) BOP-ESFAS Load Sequencer 'B' in "MISC" Mode (SAYS49)
07:44:01	Containment Normal ACU AB & CD supply temperature high (HCTS27 and HCTS28)
07:51:57.0	1st MSIS channel manually actuated (Channel 'A' – SAHS9A)
07:52:03.827	4th MSIS channel manually actuated (Channel 'D' – SAHS12A)
07:54	"Alert" EAL classification declared
07:56:21	DG 'A' Emergency Manual Trip Tripped (DGYS23) DG 'A' Status is TRIP (DGYS21) BOP-ESFAS Load Sequencer 'A' exited the "LOP" Mode
07:59:47	CEDM Exhaust Air Temperature high (HCTS57)
08:05	Aux. Operator aligns CH-E to take alternate suction from CH-327 per Standard Appendix 10A, however, he fails to open CH-327
08:05:48.5 08:05:52.2	Primary Operator (PO) Starts CH-E pump, see's flow increase to approx 80 gpm (CH-B was already running with a suction from the VCT) PO secures CH-B (flow decreases to 40 gpm)
08:05:58.9	CH-E pump trips on low suction pressure
08:06:22.2	PO restarts CH-B (suction on VCT), flow restored to 40 gpm
08:10:45.8	CH-B secured
08:11:31.0	CH-B started. Suction realigned to CH-536
09:51	Offsite Power restored to PBAS03 and Unit 2 exited the Emergency Plan (both the "Alert" and "NUE" EAL). Also at this time, Unit 3 exited the "NUE" EAL classification.
12:07	Unit 1 restored offsite power to PBBS04 and exited the "NUE" EAL classification.

Initial Areas of Concern

1. DG 'A' had no indication of voltage, current, or frequency. Suspect problem in the DG 'A' voltage regulation circuit. DG 'A' was quarantined.
2. The "A" BOP-ESFAS sequencer did not load shed the buss after the DG-A voltage regulation failure.
3. SG pressures reached 1250-1260 psia which is within 3% (tolerance) of the first set of MSSV's (setpoint 1265psia). Review of ERFDADS trends, including MSSV acoustic flow sensors, it appears that MSSV's did not lift, however, after discussion with Maintenance Engineering it is recommended that the 4 MSSV's with the lowest sepoint (1265 psia) are tested.
4. Class 1E Batteries 'A' and 'C' supplied power for 110 minutes and voltage lowered to 118 VDC.
5. CHE-P01 tripped on low suction pressure (human error). (CRDR 2716521)
6. RK problem resulted in all Control Board B05A and B RK window alarms.
7. RCP 1B Lift Oil Pump experienced an '86' lockout.

Immediate corrective Actions

- Identify and correct the cause of the and DG 'A' problem(s).
- Verify CH-E pump did not sustain any damage due to the low suction pressure trip.
- Testing of the SDC suction interlocks (SR 3.0.3) will restrain the unit in the lowest mode achieved until testing is completed.

Contributing Causes

Switchyard problem developed due to a unisolated fault on the 230KV Liberty transmission lines in the North Phoenix Valley.

Unit 2 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

General Plant Response

Plant performance for this reactor trip was generally as expected. The LOP automatic actuation occurred as expected. A manual MSIS actuation occurred as directed by the EOPs.

PLANT PROTECTION SYSTEM/ENGINEERED SAFETY FEATURES ACTUATION SYSTEM EVALUATION

I. PLANT PROTECTION SYSTEM EVALUATION

A review of the RONAN and PMS alarm typer printouts was performed in order to verify that the UFSAR Limit (Table 7.2-4AA) for reactor trip system response time on DNBR - Low. PMS shows the following:

07:41:17.385	LO DNBR CH C	TRIP	SOE
07:41:17.407	LO DNBR CH B	TRIP	SOE
07:41:17.431	LO DNBR CH A	TRIP	SOE
07:41:17.446	LO DNBR CH C	TRIP	SOE
07:41:17.489	CEDM POWER BUS UNDV 1	YES	SOE
07:41:17.490	CEDM POWER BUS UNDV 2	YES	SOE
07:41:17.490	CEDM POWER BUS UNDV 3	YES	SOE
07:41:17.493	CEDM POWER BUS UNDV 4	YES	SOE

LO DNBR CH B was received at 07:41:17.407 and CEDM POWER BUS UNDV 4 was received at 07:41:17:493 yielding a response time of 0.086 seconds. The shortest response time requirement in Table 7.2-4AA for DNBR - Low is 0.30 seconds. The 0.086 response time for this event would be well within the most restrictive for DNBR - Low. It should be noted that two HI LPD tripped channels (CH 'C' and 'B') resulted at 07:41:17.408 and this indication is shown on the Reactor First Out Panel. The difference between the DNBR and LPD trip initiation was 1ms. This explains why the first-out indicator (mechanical relay) show LPD came in first.

II. ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (NSSS & BOP-ESFAS) EVALUATION

Loss of Power (LOP) is discussed below. No other automatic ESFAS actuations were required or were initiated.

DG 'B' received its start signal at 07:41:18.9 (PTARS) and energized PBB-S04 about 5 seconds later, at 07:41:24.4 (Breaker Closed, PEYS2C-PTARS) Output Voltage > 3740V at 07:41:24.5 (5.6 sec) and DG frequency > 58.5Hz 07:41:25.3 (6.4 sec) – well within 40ST-9DG02 acceptance criterion of 10 seconds. DG 'B' maintained PBB-S04 bus load with no issues.

DG 'A' also received its start signal at 07:41:19.6 and energized PBA-S03 about 6 seconds later, at 07:41:25. This was well within the 40ST-9DG02 acceptance criterion of 10 seconds. After about 20 seconds, at approximately 07:41:45, bus voltage started to drop. In about 15 seconds, by 07:42:00, bus voltage had lowered to and remained at around 100 volts. DG 'A' was manually tripped at approximately 07:56.

Unit 2 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

U2 DG 'A' SOE

Time	Event	Source
07:41:13	ESF BUS UNDV CH A-2 (SAYS20)	PMS TYPER
07:41:15	ESF BUS UNDV CH A-4 (SAYS22)	PMS TYPER
07:41:15	LOP/LOAD SHED A (SAYS9)	PMS TYPER
07:41:15	DG START SIGNAL A (SAYS11)	PMS TYPER
07:41:19.6	DG 'A' start signal (DGYS7)	ERFDADS/PTARS
07:41:25.2	DG 'A' voltage > 3740V (PEE01)	ERFDADS/PTARS
07:41:25.7	DG 'A' frequency > 58.8 Hz (PES01)	ERFDADSPTARS
07:41:25.7	DG 'A' voltage ~4160V (PEE01)	ERFDADS/PTARS
07:41:26.0	DG-A Breaker Closed (PEYS1C)	ERFDADS/PTARS
07:41:27.5	DG 'A' voltage stable ~4200V (PEE01)	ERFDADS/PTARS
07:41:27.5	DG 'A' frequency stable~60 Hz (PES01)	ERFDADS/PTARS
07:41:28	PBAS03 voltage >4160V (PBE03)	ERFDADS
07:41:45.8	DG 'A' voltage < 4160V (PEE01)	ERFDADS/PTARS
07:41:46	PBAS03 voltage <4160V (PBE03)	ERFDADS
07:56:21	DG 'A' EMER MAN TRIP (DGYS23)	PMS TYPER
07:56.22.4	DG 'A' trip signal (DGYS7)	ERFDADS/PTARS

PLANT PERFORMANCE EVALUATION

The Plant Performance Evaluation is an overall evaluation of how the plant responded to a transient and/or subsequent reactor trip. The evaluation should assess and describe items such as equipment malfunctions, erroneous actuations, etc., that occurred which were not directly associated with the initiating event or its contributing cause(s) or associated with systems or equipment used or mitigate or recover from the actual event. A review of available information should be performed to identify:

I. SAFETY FUNCTION IMPACT

A. Reactivity Control - No impact during this event. Automatic Reactor trip was required and successful.

B. Maintenance of Vital Auxiliaries – Loss of 'A' train Class 1E 4.16KV challenged MVAC. DG-A was placed in Emergency Shutdown due to a failure to maintain output voltage 15 minutes after is started. System Engineering evaluated the fact that DG-A ran for about 15 minutes without cooling and concluded there was no impact to the Engine as no high temperature conditions were created. The BOP-ESFAS load sequencer did not load shed the breaker that were closed in since the Sequencer was only 20 seconds into the 60 second LOP cycle when the DG-A voltage regulation failed. While in the LOP cycle, a second load shed pulse is blocked for 60 seconds as the large loads are started. With this condition, all required equipment was available on the "B" train.

Unit 2 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

C. Heat Removal – The RCS achieved satisfactory Natural Circulation flow and cooled the core. RCS delta-T was approximately 34°F. T-cold reached a maximum of 574°F following the reactor trip and establishing heat removal via the Atmospheric Dump Valves. Steam Generator pressure reached a maximum of 1260 psia and a minimum of 1080 during this event. This was within 1% of the lower MSSV setpoint of 1264 psia. To prevent a high Pressurizer level, a plant cool-down was initiated and proper Pressurizer level was maintained. The restoration of forced circulation was delayed by an over-conservative Standard Appendix 1 step that requires the RCP amperage to be above 360 amps without accounting for high grid voltage conditions. If not met the action requires the RCP to be secured. The purpose of the low amp limit is to ensure that two phase flow is not present due to low subcooling margin. This ampere limit has been changed to 345 amps.

D. Pressure and Inventory Control - The maximum pressure reached during the event was 2294 psia while a minimum of 2087 psia was reached. The maximum pressure was well below the lower relief setpoint of 2500 psia. The maximum pressurizer level reached was 66% and the minimum was 26%. Pressurizer level was ultimately stabilized at 40%. Human error in the alignment of the CH-E pump suction resulted in a minor challenge to restoration of the charging system.

E. Containment Integrity –No impact during this event.

F. Containment Atmospheric Control – A loss of Containment cooling occurred and Containment temperature exceeded the Tech Spec LCO 3.6.5 limit of 117°F for 3 hours (maximum observed temperature = 123.5°F)

II. Equipment/component malfunctions

A. DG 'A' voltage regulation. DG 'A' was QUARANTINED.
B. RCP 1B Lift Oil Pump sustained an '86' lockout.
C. RKNB05A and RKNB05B

UNIT 2

SAMPLE
PLANT TRANSIENT REVIEW ASSESSMENT

PVNGS INVESTIGATION PROGRAM
PLANT TRANSIENT REVIEW ASSESSMENT

	EVENT DATE 6/14/04	EVENT TIME ~~0744~~ 0740 MST

BRIEF DESCRIPTION LOOP (UNIT 2)

SHIFT PERSONNEL

SHIFT MANAGER Dave Burns		CONTROL ROOM SUPERVISOR Bob Carbonneau
PRIMARY R.O. Lou Berberich	SECONDARY R.O. Roger Miller	THIRD R.O.
SHIFT TECHNICAL ADVISOR Jim Moreland		OTHER PERSONNEL INVOLVED IN EVENT

PLANT CONDITIONS PRIOR TO THE EVENT

MODE	REACTOR POWER	MWe	CEA POSITION	PRESSURIZER PRESSURE	TAVG
1	100		ARO	2250	587

PRESSURIZER LEVEL	BORON CONCENTRATION	S/G LEVEL (WIDE RANGE) #1	S/G LEVEL (WIDE RANGE) #2
52	992	75WR / 50(NR)	75WR / 50(NR)

EVOLUTIONS IN PROGRESS PRIOR TO THE EVENT
None

CONTROL SYSTEM STATUS PRIOR TO EVENT

REACTOR REGULATING SYSTEM: OPERATE or TEST Tavg Selected 1 2 AVG.

CEDMCS: AS MS MG MI STANDBY

FEEDWATER CONTROL SYSTEMS:

	S/G # 1	S/G # 2
MASTER:	MAN/AUTO	MAN/AUTO
DOWNCOMER REG. VALVE MANUAL/AUTO STATION:	MAN/AUTO	MAN/AUTO
ECONOMIZER REG. VALVE MANUAL/AUTO STATION:	MAN/AUTO	MAN/AUTO
FEED PUMP SPEED MANUAL/AUTO STATION:	MAN/AUTO	MAN/AUTO
BIAS SETTING:	∅	∅
GE CONTROLLER:	MAN/AUTO	MAN/AUTO

	AUTO	MAN	STPT
STEAM BYPASS CONTROL: REMOTE/LOCAL	✓		
PRESSURIZER LEVEL CONTROL REMOTE/LOCAL	✓		
PRESSURIZER PRESSURE CONTROL:	✓		

REACTOR POWER CUTBACK SYSTEM: AAOOS

RPS/ESFAS/BOP-ESFAS STATUS PRIOR TO EVENT

LIST ANY CHANNELS TRIPPED OR IN BYPASS PRIOR TO THE EVENT:

(Continued on Form PV415-04DJ)

SAMPLE
PLANT TRANSIENT REVIEW ASSESSMENT

PVNGS INVESTIGATION PROGRAM
PLANT TRANSIENT REVIEW ASSESSMENT - CONTINUED

EVENT DATE: 6/14/04
EVENT TIME: 0740 MST

REACTOR TRIP INITIATION

RECORD REACTOR 1st - Out: LPD Hi
RECORD TURBINE 1st - Out: MECH OVERSPEED

Was the Initiating Reactor Trip Parameter (if available) within the required value? ☐ YES N/A ☐ NO

If NO, Explain: Reactor Engineering to analyze the CPC Trip Time

Was the response time (i.e., the time from reaching the process setpoint until the bus undervoltage alarm occurs.) of the initiating trip signal within the required value as listed in the PVNGS Technical Specifications? ☑ YES ☐ NO

If NO, Explain:

ESFAS/BOP-ESFAS ACTUATIONS

Was Pressurizer Pressure below th SIAS setpoint (1837 psia)? ☐ YES ☑ NO

If YES, then HPSI injection has occurred. Refer to TRM 3.5.203, condition (A). Notify the Mech. System Engineering Section Leader and Mechanical Design Engineering Section Leader.

RECORD ANY ESFAS ACTUATIONS OR ESFAS-BOP ACTUATIONS AND THE CHANNELS ACTUATED.

Were any RCS pressure/temperature limits, as listed in Tech Specs violated? ☐ YES ☑ NO

If YES, provide summary and evaluation in the Incident Investigation Report. Outline the circumstances defining the occurrence, the results of any Engineering Evaluations performed to evaluate the impact on the RCS.

Does this event involve a potentially damaging transient (i.e., waterhammer event) requiring piping inspection? ☑ YES ☐ NO

The areas to be inspected are

Person(s) Contacted JONES

If further inspection is required, such as inspection of hydraulic and mechanical snubbers per Technical Requirement Manual (TRM) TSR 3.7.101.1.d, ensure appropriate tracking mechanism is initiated.

Have one or more Main Steam Safety Valves actuated? ☐ YES * ☑ NO

If YES, then declare those MSSVs INOPERABLE(refer to LCO 3.7.1) until setpoint testing verifies operability and contact System Engineering to determine needed testing of the MSSVs.

Person(s) Contacted * No evidence of lift but setpoint was approached (< 3% margin) recommended MSSV testing

Did RCS pressure exceed 2750 psia? ☐ YES ☑ NO

If YES, provide a discussion in the Incident Investigation Report which enumerates the actions taken to comply with Technical Specification 2.2.

Has any MSR pressure exceeded 248 psig? ☐ YES ☑ NO

Determine whether this occured by reviewing ERFDADS data (MTP414CP, MTP415CP, MTP416CP, & MTP417CP). Pressure in excess of 248 psig could indicate that the capacity of an MSR relief valve has been exceeded. (215 psig max)

If YES, notify the Mechanical Design Engineering Section Leader and Mechanical Maintenance Engineering Section Leader.

(STA SIGNATURE) 6/15/04 (DATE)

212-00CA Rev. 2-89 4190-900121 (BX/500)

APS

Arizona Public Service Company

COMPANY CORRESPONDENCE

ID#: 162-10934-GWA/EMF

DATE: June 15, 2004

TO: James T. Taylor
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Ext.: 82-6080

FROM: George W. Andrews
Sta.#: 7693
Ext.: 82-5709

Digitally signed by: Flodin, Erik M(Z35575)
Date: 06/18/2004 15:13:38
Reason: I have reviewed this for George Andrews
Location: PVNGS

FILE:

SUBJECT: **Revision 01 - Safety Limit Evaluation for Unit 2 Reactor Trip on June 14, 2004**

Unit 2 experienced an automatic reactor trip from 100% power, following a grid disturbance which removed electrical power from the reactor coolant pumps, on June 14, 2004 at 0741. The reactor trip was initiated by the CPC Low DNBR function on CPCs B & C due to Flow DNBR (PID406) [quickly followed by CPC A & D] which is approximately Static DNBR(PID349) multiplied by 0.1 due to RCP speed being less than 95% rated speed. Reactor Engineering has reviewed the data from PTARS and the CPC and CEAC Trip buffers and has concluded that the Safety Limits for DNBR, Peak Fuel Centerline Temperature, and Pressurizer Pressure, were not exceeded.

At the time of the reactor trip, CPCs A, B, C and D tripped in response to the low flow modification of DNBR. CPC D had been previously declared Inoperable but remained functional. Static DNBR remained larger than the Trip Setpoint on all 4 CPC Channels.

CPC A: PID 406= 0.17201 PID 349 = 1.8171

CPC B: PID 406 = 0.18140 PID 349 = 1.8747

CPC C: PID 406 = 0.17340 PID 349 = 1.8785

CPC D: PID 406 = 0.18171 PID 349 = 1.9617

The Peak Fuel Centerline Temperature SAFDL was not exceeded as evidenced by LPD values on all 4 CPC being less than 21 kW/ft.

PTARS data indicates that maximum RCS pressure during the event was approximately 2300 psia. Therefore the safety limit of 2750 psia was not challenged during this event.

If you have any questions or need further assistance, please contact Erik Flodin at extension 5899 or pager 602-226-1120.

GWA/EMF/emf

cc:	C. K. Seaman	7693	R. P. Bandera	7693
	G. W. Andrews	7693	D. W. Vogt	7833
	W. D. Chapin	7693	Reactor Eng. Route	7693
	M. Grigsby	7298	J. P. McDowell	7997

APS A subsidiary of Pinnacle West Capital Corporation

Company Correspondence

ID # 469-00368/MSC/dlh

DATE: June 18, 2004

TO: Michael Grigsby
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Ext. # 3616

Digitally signed by: Coppock, Michael S(Z00328)
Date: 06/18/2004 10:56:04
Reason: I am approving this document
Location: PVNGS

FROM: M. S. Coppock
Sta. # 7565
Ext. # 82-5990

SUBJECT: Unit 2 Grid Problem Reactor Trip of June 14, 2004 - Control Systems Response Evaluation

Event Summary

On June 14, 2004, Unit 2 was operating at approximately 100% power and had no significant evolutions in progress. At around 07:41 a grid disturbance and subsequent electrical protective actions resulted in a loss of electrical power to all 13.8KV electrical buses. As a result, all four RCPs lost power and the reactor tripped on Low DNBR.

All Non-Class power was initially lost thereby disabling the Control Systems. Most of the Control Systems are powered from Panels D11 and D12. Both D11 and D12 are supplied with Static transfer switches that will provide back-up power from the Diesel busses if available. Since the diesels were starting, the loss of D11 and D12 was momentary but long enough to disable the control systems. Power to D11 was lost for the remainder of this event when the 2A diesel was tripped. The MFWPTs controls, one source of EHC power and other miscellaneous controls are powered from D15 and D16.

The response of the Control Systems (FWCS, SBCS, RPCS & RRS) are not required to operate for PVNGS Nuclear Safety per UFSAR Chapter 7.7/CESSAR 7.

Feedwater Control System (FWCS) Response

The System was disabled by the loss of power. The Feedwater Pumps tripped on the loss of power. Both the Economizer valves and the Downcomer valves shut with the loss of power. Auxiliary Feedwater was manually initiated several minutes into the event.

Steam Bypass Control System (SBCS) Response

The System was disabled by the loss of power.

Reactor Regulating System (RRS) Response

The System is not required for a reactor trip. This System also lost power.

PCL XL error

Subsystem: TEXT

Error: InsufficientMemory

Operator: Text

Position: 7689