Attachment C

CRDR 2715727 SWMS Attached Media for Unit 1 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 14:05:13
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Location: PVNGS

C/13

APS

DATE: June 16, 2004

Company Correspondence

Memo# 294-01899-DWV
TO: Jim McDowell
Sta. # 7997
Ext. # 5668

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

Digitally signed by: Vogt, Donald W(Z96416)
Date: 06/16/2004 18:18:59
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Location: PVNGS

SUBJECT: Unit 1 Loss of Off-Site Power, Main Generator Trip, Reactor Trip on Low DNBR of June 14, 2004.

The Shift Technical Advisors have conducted a Plant Performance of the Unit 1 Loss of Off-Site Power and subsequent Reactor Trip on 06/14/2004.

The Plant Performance Evaluation is an overall evaluation of how the plant responded to the Loss of Off-Site Power and 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), operating procedures, and personnel statements. The summary of these evaluations is provided as attachments.

Attachments
Event Summary/General Plant Response
Plant Performance Evaluation/ 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 1 Loss of Offsite Power and Reactor Trip from 99% Power

06/14/2004

EVENT SUMMARY DOCUMENT

Event Description

Unit 1 was operating steady state at 99% 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. The 13.8 KV buses (NAN-S01 and NAN-S02) received a load shed command; all four RCPs lost power resulting in reactor trip on Low DNBR (CPC penalty factor to PPS based on RCS low flow). The turbine tripped on the ETSV trip. Each DG received a start signal and powered up its respective 4.16 KV bus. The CR staff completed the SPTAs, the CRS diagnosed an LOOP/LOFC and entered 40EP-9EO07. The crew initiated manual MSIS per the EOP. This electrical distribution condition met the threshold for a Notification of Unusual Event classification and the NUE was declared at 07:58. The crew cross-tied EW-B to NC before securing the last Charging Pump. Post-trip, the Unit 1 operating crew identified that loss of NC flow had not isolated letdown, and manually closed CHBUV523 from B03. The NUE was terminated at 12:07 following restoration of off-site power. Evaluations will be documented in CRDR 2715727.

Initial Sequence of Events (non SOE times are approximate)

Time	Event - LOOP, Main Generator Trip, Reactor Trip (6/14/04)
07:40:57	GEN NEG Sequence (MAYS14)
07:40:58	EAST Bus 525 KV SWYD Volts LO-LO (MAYS90)
07:40:59	WEST Bus 525 KV SWYD Volts LO-LO (MAYS92)
07:41:17	13.8 KV SWGR 1 and SWGR 2 Load Shed
07:41:17	RCP 1B and 2B Not Operating (RCYS14 and RCYS16)
07:41:17.000	Fast Closing of IVs commanded (MTYS83)
07:41:17.347	No ETSV Pressure Trip – Main Turbine Trip (MTYS47)
07:41:17.446	LO DNBR Channel C Trip (SBTC04)
07:41:17.449	LO DNBR Channel B Trip (SBTB04)
07:41:17.452	Mechanical Overspeed Trip (MTYS16)
07:41:17.506	CH B Trip Circuit Breaker Open (SBMTB10)
07:41:17.512	CH C Trip Circuit Breaker Open (SBMTC10)
07:41:17.656	The 4th CEDM Power Bus Undervoltage Alarm (SFCE9UV1-4)
07:41:21 approx	Unit 1 Generator Output Breaker Trip (From SRP SOE)
07:41:43	4160 SWGR S03 bus voltage normal
07:41:44	4160 SWGR S04 bus voltage normal
07:51:09	MSIS Manual Actuation per 40EP-9EO07
07:54	Alert declared for Unit 2 condition
07:58	NOUE declared for Unit 1
08:40	ENS Notification completed to NRC
09:00	CHBUV523 Manually Closed due to High Temperature in Letdown System and failure of CHNTSH224 to isolate letdown on 148 degrees F.
09:39	Charging Secured due to Pressurizer level approaching 56%.
12:07	Unit 1 NOUE terminated

Unit 1 Loss of Off-Site Power and Reactor Trip from 99% Power

06/14/2004

Initial Areas of Concern

1. Nuclear Cooling Water Flow Low Interlock on CHBUV523 was defeated by T-Mod.
2. CHNTSH224 failed to isolate Backpressure Control Valves at 148 ° F.
3. High Temperature water diverted to CVCS Hold Up Tank. Letdown issues addressed by CRDR 2715667.
4. Atmospheric Dump Valve 185 apparently drifted Closed.
5. Reactor Coolant Pump Seal status. Mech. Maint. Engineering to evaluate.
6. Auxiliary Boiler Out Of Service.
7. Shutdown Cooling Pressure Interlock testing (SR 3.0.3).

Immediate corrective Actions

Identify and correct letdown system issues per CRDR 2715667.

Contributing Causes

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

General Plant Response

Plant performance for this reactor trip was generally as expected. The LOP automatic actuation occurred as expected. A manual MSIS actuation was initiated 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:

07:41:17.446	LO DNBR CH C	TRIP	SOE
07:41:17.449	LO DNBR CH B	TRIP	SOE
07:41:17.457	LO DNBR CH D	TRIP	SOE
07:41:17.471	LO DNBR CH A	TRIP	SOE
07:41:17.526	CEDM POWER BUS UNDV 2	YES	SOE
07:41:17.605	CEDM POWER BUS UNDV 1	YES	SOE
07:41:17.622	CEDM POWER BUS UNDV 3	YES	SOE
07:41:17.656	CEDM POWER BUS UNDV 4	YES	SOE

LO DNBR CH B was received at 07:41:17.449 and CEDM POWER BUS UNDV 4 was received at 07:41:17:656 yielding a response time of 0.207 seconds. The shortest response time requirement in Table 7.2-4AA for DNBR - Low is 0.30 seconds. The 0.207 response time for this event is within the most restrictive for DNBR - Low.

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

Loss of Power (LOP) was received due to the loss of off-site power. After start signal, DG 'A' restored PBB-S03 bus voltage >3740V in approximately 5.7 seconds, and frequency > 58.8 Hz in approximately 6.7 seconds. After start signal, DG 'B' restored PBB-S04 bus voltage > 3740 in approximately 5.7 seconds and frequency > 58.8 Hz in approximately 6.5 seconds. These times are well within the 40ST-9DG01/02 acceptance criterion of 10 seconds.

There were no other automatic ESFAS actuations required or initiated.

PLANT PERFORMANCE EVALUATION

I. SAFETY FUNCTION IMPACT

A. Reactivity Control - Automatic Reactor trip was required and successful.

B. Maintenance of Vital Auxiliaries – Non-Class 1E power was lost during this event. After LOP, each Class 1E bus was supplied from its respective diesel generator.

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 570°F following the reactor trip and establishing heat removal via the Atmospheric Dump Valves. To prevent a high Pressurizer level several hours post-trip, charging flow was secured and proper Pressurizer level was maintained. SG#1 pressure reached a maximum of approximately 1190 psia and a minimum level of approximately 40%WR. SG#2 pressure reached a maximum of approximately 1190 psia and a minimum level of approximately 46%WR.

D. Pressure and Inventory Control - No impact during this event. The maximum pressure reached during the event was 2285 psia while a minimum of 2100 psia was reached. The maximum pressurizer level reached was 53% (normal level at full power) and the minimum was 45%. Pressurizer level was ultimately stabilized at approximately 48%.

E. Containment Integrity - No impact during this event.

F. Containment Atmospheric Control - A loss of Containment cooling occurred and Containment temperature rose to approximately 120F until EW was cross-tied to NC and chilled water restored to containment (LOOP/LOFC safety function status check acceptance criterion is < 117F). CRDR 2715941 was initiated to address EQ program impact. Transient analysis includes an evaluation for impact on DBA.

II. Equipment/component malfunctions

A. CHNTSH224 failure to isolate backpressure control valves was an initial concern. However, as power was cycled to the controller it returned in Manual. The temperature trip is not active in the Manual Mode of operation per design.

UNIT 1

SAMPLE
PLANT TRANSIENT REVIEW ASSESSMENT

PVNGS INVESTIGATION PROGRAM
PLANT TRANSIENT REVIEW ASSESSMENT

EVENT DATE	EVENT TIME
6/14/2004	0743

BRIEF DESCRIPTION: Loss of Offsite Pwr / Loss of Forced Circ

SHIFT PERSONNEL

SHIFT MANAGER	CONTROL ROOM SUPERVISOR	
Mike Sanchez	Larry Wherry	
PRIMARY R.O.	**SECONDARY R.O.**	**THIRD R.O.**
Dan Amour	Brian Emerzl/Kar	
SHIFT TECHNICAL ADVISOR	**OTHER PERSONNEL INVOLVED IN EVENT**	
Tim Gaffney		

PLANT CONDITIONS PRIOR TO THE EVENT

MODE	REACTOR POWER	MWe	CEA POSITION	PRESSURIZER PRESSURE	TAVG
1	99.99	1300	ARO	2250	582

PRESSURIZER LEVEL	BORON CONCENTRATION	S/G LEVEL (WIDE RANGE) #1	S/G LEVEL (WIDE RANGE) #2
52	1408	75%	75%

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:	-500	-500
GE CONTROLLER:	MAN/AUTO ☐	MAN/AUTO ☐

STEAM BYPASS CONTROL: REMOTE/LOCAL	☑ AUTO	☐ MAN	☐ STPT	PRESSURIZER LEVEL CONTROL REMOTE/(LOCAL)	☑ AUTO	☐ MAN	☐ STPT
PRESSURIZER PRESSURE CONTROL:	☑ AUTO	☐ MAN	☐ STPT	REACTOR POWER CUTBACK SYSTEM:	☐ AAOOS		

RPS/ESFAS/BOP-ESFAS STATUS PRIOR TO EVENT

LIST ANY CHANNELS TRIPPED OR IN BYPASS PRIOR TO THE EVENT:
NONE

(Continued on Form PV415-04DJ)

SAMPLE
PLANT TRANSIENT REVIEW ASSESSMENT

PVNGS INVESTIGATION PROGRAM
PLANT TRANSIENT REVIEW ASSESSMENT - CONTINUED

EVENT DATE	EVENT TIME
6/14/04	0743

REACTOR TRIP INITIATION

RECORD REACTOR 1st - Out: DNBR LO RECORD TURBINE 1st - Out: NO ETSO Press Trip

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

If NO, Explain: CPC initiated trip. Reactor Engineering will analyze

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.

LOP actuated on PBA503 and PBD504

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 MSSS / TURB BLDG

Person(s) Contacted Winsor 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

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. ~ 230 psig

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

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

PV419-04DJ (2-98)

212-00CA Rev. 2-89 4190-900121 (8X/500)

APS
Arizona Public Service Company
COMPANY CORRESPONDENCE

ID#: 162-10919-GWA/RLS

DATE: June 16, 2004

TO: James T. Taylor
Sta.#: 7848
Ext.: 82-6080

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

Digitally signed by: Andrews, George W(Z99748)
Date: 06/16/2004 10:24:08
Reason: I am approving this document
Location: PVNGS

FILE:

SUBJECT: **Safety Limit Evaluation for Unit 1 Reactor Trip on June 14, 2004**

Unit 1 experienced an automatic reactor trip from 100% power, following a grid disturbance which impacted electrical power to the reactor coolant pumps, on June 14, 2004 at 0741, resulting in a decrease in RCP speed. The reactor trip was initiated by the CPC Low DNBR function due to CPCs B and C on Flow DNBR (PID 106) [quickly followed by CPC A & D] which is approximately Static DNBR(PID 334) 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. Static DNBR was still larger than the Trip Setpoint on all 4 CPC Channels; therefore, the DNBR Safety Limit was not exceeded.

CPC A: PID 106 = 0.17558, PID 334 = 1.8159

CPC B: PID 106 = 0.17797, PID 334 = 1.8690

CPC C: PID 106 = 0.17948, PID 334 = 1.8866

CPC D: PID 106 = 0.17247, PID 334 = 1.7784

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

PTARS data indicates that maximum pressurizer pressure during the event was approximately 2285 psia. Therefore the RCS pressure Safety Limit of 2750 psia was not challenged during this event.

If you have any questions or need further assistance, please contact Bob Simmons at extension 5928 or pager 602-226-0864.

GWA/RLS/rls

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

A subsidiary of Pinnacle West Capital Corporation
COMPANY CORRESPONDENCE

ID # 469-00367/MSC

DATE: June 16, 2004

TO: Mark McGhee
Sta. # 7918
Ext. # 1088

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

SUBJECT: Unit 1 Grid Problem Reactor Trip of June 14, 2004 - Control Systems Response Evaluation CRDR 2715727

Event Summary
On June 14, 2004, Unit 1 was operating at approximately 99.9% 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. A Notification of Unusual Event was declared. A manual MSIS was initiated as directed by the EOPs approximately 10 minutes after the trip.

All Non-Class power was lost thereby disabling the Control Systems. The Control Systems are powered from Panels D11 and D12.

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.

Reactor Power Cutback System (RPCS) Response
The System is not required for a reactor trip. This System also lost power.

Pressurizer Level Control System (PLCS) and
Pressurizer Pressure Control System (PPCS) Response
Due to the loss of power to controls, the Pressurizer level and pressure was manually controlled during this event.

Should any questions arise, please contact Gary Anderson at extension 5742.

MSC/mlh/gta

cc:

C. D. Mauldin	7605
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T. L. Radtke	7294
M. J. Winsor	7669
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M. A. Radspinner	7526
G. T. Anderson	7535
D. L. Holland	7590
D. Fisher	7357

APS

Arizona Public Service Company

COMPANY CORRESPONDENCE

ID #: 162-10922-AMT

DATE: June 15, 2004

TO: James Mcdowell
Sta. #: 7997
Ext. #: 82-5668

FROM: Arshad M Taufiq
Sta. #: 7693
Ext. #: 82-6607

Digitally signed by: Taufiq, Arshad M(Z93839)
Date: 06/15/2004 15:46:50
Reason: I am the author of this document
Location: PVNGS

Nuclear Fuel Management
Transient Analysis

SUBJECT: **NFM Nuclear Safety Assessment: Unit 1, Loss of Power on June 14, 2004**

Procedure 90DP-0IP06 (Reactor Trip Investigation) requires a nuclear safety assessment to be performed which should address the affect of the event on nuclear safety, and should include a comparison of observed values to those maximum and minimum values specified in the Technical Specifications and the Safety Analysis Report. The purpose of the procedure is to complete an accurate investigation of unplanned reactor trip events. This procedure also applies to Unit restart authorization activities following reactor trip events.

Conclusion

The event experienced by Unit 1 on June 14 2004, did not result in a transient more severe than those already analyzed in the Chapter 15 of PVNGS UFSAR. The reactor tripped from 100% power on a CPC generated reactor trip on low DNBR due to low RCP speed. The event initiator was a loss of load due to grid disturbance. The response time for the reactor trip on low DNBR due to low RCP shaft speed was well within the 0.3 second response time assumed in the UFSAR Table 7.2-4AA.

The post trip NSSS response was normal with the exception of letdown flow. Letdown flow is normally isolated automatically due to loss of NCW flow during of loss of offsite power. Letdown flow and charging pumps were however secured manually which maintained adequate pressurizer level.

Equipment and systems assumed in UFSAR Chapter 15 were functional and performed as required. Scenarios defined in UFSAR Chapter 15 concerning Loss of Reactor Coolant Flow remained bounding for this event.

Containment temperature exceeded 117 °F by few degrees for a duration of 2-3 hrs post trip, due to loss of containment cooling. This did not have any negative impact on the Non-LOCA or LOCA Chapter 15 safety analysis.

For further information related to this assessment please see below or call Arshad Taufiq at extension 82-6607.

cc:

C.K. Seaman (7693)	R.P. Bandera (7693)	A. R. Fluegge (7995)	G.W. Andrews (7693)
D.W. Vogt (7833)	C.A. Hasson(7693)	J.T. Taylor(7448)	M.A. McGhee(7198)

Purpose

The primary purpose of the Safety Analysis Assessment is to address the impact of this event on nuclear safety, including automatic actuations, equipment failures, and personnel response. This assessment will include a comparison of the actual observed values to those maximum and minimum values specified in the Tech Specs and the Safety Analysis Report. One way to answer this question is to compare the event to similar analyzed events of the appropriate frequency category and ensure that the consequences of the actual event are bounded by the event analyzed. If the event is not similar to events previously analyzed, some specific analyses may be required to demonstrate the acceptance criteria are met.

General Event Description

1) The event experienced by Unit 1 on June 14, 2004 did not result in a transient more severe than those already analyzed in Chapter 15 of the PVNGS UFSAR. This event is characterized as a Loss of Reactor Coolant Flow event. The licensing basis event is presented in UFSAR 15.3.1, "Total Loss of Reactor Coolant Flow." The reactor tripped from 100% power due to a CPC generated reactor trip on low DNBR caused by low RCP pump speed. The event initiator was a grid disturbance resulting in Main Generator trip which ultimately caused loss of power to NAN-S01 and NAN-S02, the source of power to the RCP's. This resulted in reactor trip on low DNBR. Based on timing of CEDM Under voltage Coil, the response time for the reactor trip on low DNBR due to low RCP shaft speed was well within the 0.3 second response time specified UFSAR Table 7.2-4AA.

2) The post trip NSSS response was normal with the exception of letdown flow. Letdown flow which is normally isolated automatically due to loss of NCW flow was still in service due to installation of T-mod. However, letdown and charging was manually isolated thereby control of pressurizer level was maintained.

3) Equipment and systems assumed in UFSAR Chapter 15 were functional and performed as required. Scenarios defined in UFSAR Chapter 15 concerning Loss of Reactor Coolant Flow remained bounding for this event.

Evaluation

As shown on Table 1 there are three major criteria for events included in the UFSAR Chapter 15 Safety Analysis. The appropriate criteria must be used based on the frequency of occurrence of a given event. This event is classified as a reactor trip on Loss of Reactor Coolant Flow in the Anticipated Operational Occurrence (AOO) category. Events of this type are expected to occur during a calendar year. As such, this event falls into the Moderate Frequency category, and must meet the design criteria as specified on Table 1. This class of event is normally evaluated for its potential for primary and secondary pressure peaking and fuel failure. In this instance, the design criteria of interest for fuel failure is the DNBR SAFDL.

Each of the three general criteria will be specifically addressed in the following sections.

Shutdown Margin

This event did not challenge shutdown margin criteria. All CEAs inserted as designed. A loss of reactor coolant flow event is a mild heatup event and the cold leg temperatures rose slightly above normal temperature before reactor trip. Since the unit is operating with negative moderator temperature conditions, adequate shutdown margin was available throughout the event. From the ERFDADS plots the cold leg temperatures reached a maximum of 570 °F following

the reactor trip.

Peak Pressure

No postulated event should be severe enough to cause a reactor system pressure boundary rupture. The allowable peak pressure for an event is based on the event category. The event category is tied to ASME Code criteria which generally define pressure limits in terms of a percentage of the design pressure. An AOO would allow a pressure up to 110 percent of the design pressure. The ERFDADS data show that both primary and secondary pressures were well below 110% of the design pressures.The pressurizer pressure peaked at approx. 2285 psia psia. This would result in RCS pressure being well below 2750 psia (110% of primary design pressure). The pressurizer pressure before the trip was about 2250 psia. The peak secondary pressure was approximately 1170 psia (from the ERFDADS), well below 1398 psia (110% of secondary design pressure). No PSVs or MSSVs lifted as a consequence of this event.

Fuel Failure and Offsite Dose

No fuel failure occurred since the Specified Acceptable Fuel Design Limit (SAFDL) Departure from Nucleate Boiling Ratio (DNBR) was not exceeded during the event.

Conclusions

The conclusion of this assessment is that the Reactor Trips experienced by Unit 1 on June 14, 2004 did not result in a transient more severe than those already analyzed. The primary system and secondary pressure boundary limits were not approached. The transient did not cause any violation of the SAFDLs.

Finally, equipment and systems performance remained consistent with that assumed in the Safety Analysis. Plant response was normal for the situation that occurred. Scenarios defined in UFSAR Chapter 15 and design assumptions of the reactor protection system will remain bounding for this reactor trip. Scenarios defined in UFSAR Chapter 6, concerning Loss Of Coolant Accidents (LOCA), were not applicable to this transient.

Reference

1.PVNGS UFSAR, 15.3.1, "Total Loss of Reactor Coolant Flow.

CATEGORY	Anticipated Operational Occurrences (AOO)		LIMITING FAULTS
	MODERATE FREQUENCY	INFREQUENT INCIDENTS	
Frequency	May occur during a calendar year.	May occur during a plant lifetime	Low probability of occurrence during a plant lifetime
Assumptions	Does not assume a Single Failure	Assumes most limiting Single Failure	Assumes most limiting Single Failure
Criteria	Shutdown Margin is greater than zero (Modes 2 -6) DNBR greater than SAFDL (i.e., no fuel failure allowed) Peak Pressure is less than 110% of Design Pressure (reference 3)	Shutdown Margin is greater than zero (Modes 2 -6) DNBR can violate SAFDL (i.e., small amount of fuel failure is permitted) and Dose Consequences are limited to a small fraction of 10CFR100 requirements. Peak Pressure is less than 110% of Design Pressure (reference 3)	Shutdown Margin is greater than zero (Modes 2 -6) DNBR can violate SAFDL (i.e., fuel failure is permitted, with core coolability maintained) and Dose Consequences are limited to less than 10CFR100 requirements. Peak Pressure is less than 110%* of Design Pressure (reference 3)

* - Except for Large Feedwater Line Break and CEA Ejection events which allow 120% of design pressure

The classification of moderate frequency, infrequent incidents and limiting faults are described in ANSI 18.2 (Reference 1 below). Standard Review Plan (Reference 2 below) discusses the requirement for moderate frequency events. The definition in the SRP and ANSI are the same. These categories are summarized above:

<u>References:</u>

1. ANSI N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactors," American National Standards Institute (1974).
2. NUREG-75-087, "Standard Review Plan for the Review of Safety Analyses Reports for Nuclear Power Plants, LWR Edition", September 1975, USNRC.
3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Over-Pressure," American Society of Mechanical Engineers.