REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8701070351 DOC. DATE: 86/12/26 NOTARIZED: NO DOCKET #
FACIL: STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529
AUTH. NAME AUTHOR AFFILIATION
HAYNES, J. G. Arizona Nuclear Power Project (formerly Arizona Public Serv
RECIP. NAME RECIPIENT AFFILIATION
KNIGHTON, G. W. PWR Project Directorate 7

SUBJECT: Forwards response to operational concerns discussed during 861028 meeting. Util actively pursuing both short & long term mods to core protection calculators (CPC) which will allow CPC to ride through transients w/o unnecessary trips.

DISTRIBUTION CODE: A001D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 31
TITLE: OR Submittal: General Distribution

NOTES: Standardized plant. M. Davis, NRR: 1Cy. 05000528
Standardized plant. M. Davis, NRR: 1Cy. 05000529

	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME		COPIES LTTR	COPIES ENCL
	PWR-B EB		1	1	PWR-B PEICSB		2	2
	PWR-B FOB		1	1	PWR-B PD7 LA		1	0
	PWR-B PD7 PD	01	5	5	LICITRA, E		1	1
	PWR-B PEICSB		1	1	PWR-B RSB		1	1
INTERNAL:	ACRS	09	6	6	ADM/LFMB		1	0
	ELD/HDS3		1	0	NRR/DHFT/TSCB		1	1
	NRR/ORAS		1	0	REG FILE	04	1	1
	RGN5		1	1				
EXTERNAL:	EG&G BRUSKE, S		1	1	LPDR	03	1	1
	NRC PDR	02	1	1	NSIC	05	1	1
NOTES:			1	1				

TOTAL NUMBER OF COPIES REQUIRED: LTTR 30 ENCL 26

Arizona Nuclear Power Project
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

December 26, 1986
ANPP-39506-JGH/BJA/98.05

Director of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Project Director
PWR Project Directorate #7
Division of Pressurized Water Reactor Licensing-B
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1 and 2
Docket Nos.: STN 50-528 (License No. NPF-41)
STN 50-529 (License No. NPF-51)
ANPP Response to Operational Concerns
File: 86-E-056-026; 86-F-056-026

Reference: (1) Meeting between ANPP and NRC personnel on October 28, 1986.
Subject: Palo Verde Management Meeting.

Dear Mr. Knighton:

During the October 28, 1986 meeting between ANPP and the NRC Staff, ANPP committed to provide the NRC Staff with a formal response to each of the concerns that were discussed at the meeting. Attachment 1 provides a detailed summary of each item.

If you have any additional questions on this matter, please contact Mr. W. F. Quinn of my staff.

Very truly yours,

J. G. Haynes
Vice President
Nuclear Production

JGH/BJA/jle
Attachment

cc: O. M. De Michele (all w/a)
E. E. Van Brunt, Jr.
E. A. Licitra
R. P. Zimmerman
A. C. Gehr

A001
1/1

OCTOBER 28, 1986 ITEM SUMMARY

1. CONCERN

The Core Protection Calculator (CPC) Departure from Nucleate Boiling Ratio (DNBR) program needs to be modified to eliminate the flow projected DNBR trip. When a fast bus transfer occurs, a 7 rpm decrease in Reactor Coolant Pump (RCP) speed is sufficient to cause a flow projected DNBR trip to occur. This appears to be a problem that the CPC DNBR flow projection algorithm is too sensitive to small underfrequency events.

ANPP RESPONSE

On September 11, 1986, with PVNGS Unit 2 at 99% reactor power, a large load rejection test was initiated by simulating a generator differential trip. The expected results of the test were the tripping of the turbine/generator, a successful fast bus transfer, and a reactor power reduction by the Reactor Power Cutback System (RPCS). When the test was initiated, the turbine/generator was tripped and a successful fast bus transfer occurred. During the generator trip/fast bus transfer, the RCP's experienced an average speed reduction due to bus underfrequency of approximately 7 rpm. This RCP speed reduction was enough to cause the CPC's to generate a flow projected low DNBR reactor trip. In the interim, PVNGS Units 1 and 2 are currently operating on the startup transformers in order to remove the need for a fast bus transfer.

ANPP is actively pursuing both short and long term modifications to the CPCs which will allow the CPCs to ride through normal transients such as generator trips without causing unnecessary reactor trips. The CPC modifications are a part of the CPC Improvement Program that ANPP has participated in along with the other utilities that have CPC plants. The methodology of this program has been reviewed and approved by the NRC Staff as part of their review of topical reports CEN-308-P and CEN-310-P.

Implementation of the CPC Improvement Program at PVNGS should prevent unnecessary reactor trips such as the one that occurred at PVNGS Unit 2 on September 11, 1986. One of the key elements of the CPC Improvement Program is the removal of the flow projected low DNBR trip. The flow projected low DNBR trip will be replaced by a CPC trip that will occur when the speed of one or more of the RCPs drops below a fixed trip setpoint on RCP speed. This CPC change can be implemented within the current safety analyses. The current safety analyses assume that a reactor trip is generated by the CPCs within a certain time following event initiation. Upon removal of the flow projected low DNBR trip, the integrity of the existing safety analyses is ensured by selection of the RCP pump speed trip setpoint such that a reactor trip is generated within the same time frame following event initiation.

ANPP currently plans to implement the CPC Improvement Program in the planned software update for Cycle 2 for each of the PVNGS units. However, ANPP is currently investigating the possibility of implementing the removal of the flow projection and other parts of the CPC improvement program at an earlier date prior to Cycle 2. The short term removal of the flow projection trip would not involve the actual removal of the algorithm from the computer but it would modify the constants such that the flow projection program would not generate a trip. CE is currently determining the lowest fixed flow trip setpoint that would replace the flow projection trip at PVNGS. Further evaluation by ANPP is necessary after CE provides the fixed flow setpoint before the change can be implemented.

2. CONCERN

The Steam Bypass Control System responds as expected when actuated from power levels above 70%. However, when power level is below 70%, the system has a tendency to over respond. This appears to be a fine tuning problem, and Combustion Engineering is involved in providing a solution.

ANPP RESPONSE

Satisfactory operation of the Steam Bypass Control System (SBCS) during load rejection and loss of feedpump events has been demonstrated at PVNGS by successful completion of planned transient testing at 50%, 70%, 80% and 100% power, as well as an unplanned turbine trip at 50% power and an unplanned feedwater pump trip at 100% power. However, there have been several reactor trips between the 25% to 40% reactor power level where a rapid rise in Steam Generator pressure could not be overcome by modulation of the steam bypass valves, and a reactor trip occurred due to high pressurizer pressure.

As a result of these trips, the SBCS was the subject of an engineering evaluation resulting from concerns that the system may not have the capability to prevent a reactor trip following a load rejection in the range of 25% to 40% reactor power. The result of this evaluation was that the SBCS operated as intended but equipment unavailability (all eight bypass valves not available) and procedure inadequacies (procedures have been corrected) prevented the SBCS from sufficiently reducing secondary pressure.

Combustion Engineering has recently stated that their computer simulations predict that the SBCS will perform at all power levels, per design, in the automatic mode, if full bypass capacity and supporting systems capability is available. CE proposed that overall SBCS performance could be enhanced by a fine tuning for the as-built steam pressure header configuration. In addition, by taking other factors into consideration, a more acceptable bypass valve response scheme could be developed (e.g., a combination of bypass valve quick open and modulation at low power levels). The CE proposal for enhancing the SBCS was determined to be unnecessary at this time due to the evaluations which show that SBCS response is adequate if all equipment is available.

3. CONCERN

Reactor Coolant Pump (RCP) coastdown is quicker than indicated in the safety analysis. A re-analysis was completed indicating that the existing flow conditions are within the bounds of the current safety analysis.

ANPP RESPONSE

During the power ascension testing program at PVNGS Unit 1, a turbine trip from 100% reactor power was initiated on January 9, 1986. Following the turbine/generator trip, an unsuccessful fast bus transfer resulted in a loss of power to the RCPs and the reactor tripped on flow projected low DNBR. During the evaluation of the data from this reactor trip, it was noted that the RCPs coasted down at a rate that was slightly faster than the rate assumed in the safety analyses. Additionally, during the post-core hot functional testing of PVNGS Unit 2, RCP coastdown rates that were slightly faster than the safety analysis assumptions were observed.

After observing these RCP coastdown rates, penalty factors were applied to the CPCs and the Core Operating Limit Supervisory System (COLSS) in PVNGS Units 1 and 2 to account for this non-conservatism. Subsequent analysis by the NSSS vendor justified the removal of these penalty factors. The re-analysis took into account the fact that the CPCs generate an ealier reactor trip than what was assumed in the safety analysis. This earlier reactor trip leads to results that are less severe than the present accident analysis because the earlier reactor trip compensates for the faster coastdown rate.

For additional information on this subject please refer to the following previously documented correspondence between the NRC and ANPP:

i) Letter from E. E. Van Brunt, Jr., ANPP, to USNRC Document Control Desk, dated February 10, 1986 (ANPP-34972). Subject: Licensee Event Report 86-006-00.

ii) Letter from E. A. Licitra, NRC, to E. E. Van Brunt, Jr., ANPP dated July 8, 1986. Subject: Request for Additional Information - Palo Verde Unit 1 LER No. 86-006.

iii) Letter from J. G. Haynes, ANPP, to G. W. Knighton, NRC, dated July 29, 1986 (ANPP-37623). Subject: NRC Request for Additional Information on the January 9, 1986, Reactor Trip at PVNGS Unit 1.

4. <u>CONCERN</u>

During a partial loss of non-Class 1E power, the motor operated isolation valves in the main steam supply to the moisture separator reheaters remained open due to the loss of power. Simultaneously, the high level dump valves on the reheater drain tanks opened to the condenser, creating a path for steam flow from the main steam system to the condenser. This in turn caused the pressure in the steam generator to decrease sufficiently to actuate the Main Steam Isolation Valves (MSIV's). Design changes will be implemented to correct the problem.

<u>ANPP RESPONSE</u>

On July 12, 1986, PVNGS Unit 1 was operating at approximately 100% of rated thermal power when a reactor trip occurred. The event was further complicated by a Safety Injection Actuation Signal (SIAS) and a Main Steam Isolation Signal (MSIS). These two signals were the direct result of the overcooling of the primary system due to an excessive steam demand after the reactor/turbine trip.

The excessive steam demand event was caused by the motor operated steam source valves on the Moisture Separator Reheater (MSR) 2nd stage steam supply lines remaining in the open position when their power supply was lost (due to the NAN-S01 and S02 load shed) and the 2nd stage reheater drain tank high level dump valves to condenser opening (as designed) on a turbine trip (below 20% turbine power). Thus, these open valves created four direct flow paths for main steam to the condenser. Refer to the attached figure for a simplified drawing of this system (note that this figure encompasses only one of the four MSR's).

A design modification has been implemented in PVNGS Units 1 and 2 to ensure that these steam flow paths are isolated to prevent future overcooling events. The modification involves the addition of AC power relays which will sense the availability of power to the motor operated steam source valves (UV-328A on figure). If power is lost to the steam source valves, then the 2nd stage reheater drain tank high level dump valves to condenser (LV-823 on figure) will be prevented from opening. Additionally, the 2nd stage reheater drain tank scavenging steam line vent valve to condenser (FV-711B on figure) will be prevented from opening on loss of power to the steam source valves and a turbine trip. These modifications improve the safety of the plant by reducing the likelihood of experiencing overcooling events following trips.

5. CONCERN

The Balance of Plant (BOP) Engineered Safety Features Actuation System (ESFAS) and Radiation Monitoring System (RMS) equipment have caused numerous actuations of the ESF systems. The problems have been varied and have included inadequate ventilation in cabinets, bad connections, and inadequate grounding.

ANPP RESPONSE

The specific problems associated with the BOP ESFAS and RMS which have caused ESF actuations have been fixed or are currently being resolved. These problems have been reported and described in various Licensee Event Reports submitted to the NRC. Specific resolutions to the items cited in the concern above are described below:

(1) Inadequate Ventilation in BOP ESFAS cabinet - On December 16, 1985, Unit 1 was operating at 52% reactor power when an electronics failure in the Train "A" BOP ESFAS cabinet resulted in the spurious actuation of several ESF signals. The cause of the event was traced to failure of a temporary fan in the BOP ESFAS cabinet. Failure of the fan allowed the ESF load sequencer module to overheat and malfunction. This problem has been resolved by replacing the temporary fan inside the cabinet with redundant, permanent, cooling fans. Also, alarms in the control room which will annunciate on high air temperature inside the BOP ESFAS cabinets have been installed. The sequencer module was also replaced.

(2) Bad Connections - BOP ESFAS - On April 28, 1986, Unit 2 was in Mode 3 when an inadvertent BOP ESFAS Train "A" actuation signal was initiated. A quality assurance inspector and an instrumentation and controls technician were inspecting the back panel of the BOP ESFAS cabinet as part of a work order. Some wires were moved for better visibility and this movement initiated the ESF actuation. It was discovered that all the male pin connectors utilized in the BOP ESFAS cabinets were not fully inserted into the female connector block. Also, the wires crimped into the female connector were found to have broken strands. The cause of the ESF actuation was attributed to inadequate pin/socket contact. In order to correct this problem, all ESFAS modules have been repinned in Unit 1, 2 and 3. Also, the female connectors in the BOP ESFAS cabinets have all been replaced and strain reliefs were installed to prohibit wire movement. These problems are further described in Licensee Event Reports submitted to the NRC on May 28 and 29, 1986 and in Deficiency Evaluation Report 86-19.

(3) Inadequate Grounding - RMS - Several Licensee Event Reports (LER's) have been submitted to the NRC concerning problems associated with the RMS for Units 1 and 2. The problems concern spurious signals sent from radiation monitors which cause an ESFAS actuation. It has been determined that many of these problems were the result of inadequate grounding of the RMS. In order to solve this problem, ANPP is installing a separate isolated grounding system for the RMS in Unit 1. Once verification is made in Unit 1 that this grounding system solves the spurious signal problem, the modification will be implemented in Units 2 and 3.

6. CONCERN

Several problems have occurred with the charging pumps. Gas binding has occurred due to low VCT level and a ruptured diaphragm in a pulsation dampener. Additionally, a crack was discovered in a charging pump block. The charging pumps impact the operability of the auxiliary pressurizer spray system.

ANPP RESPONSE

On September 12, 1985, an event occurred at PVNGS Unit 1 which resulted in the loss of charging flow for a period of time. There were a number of things that occurred during the event which contributed to the loss of charging flow. These contributing factors are identified as the failure of the Volume Control Tank (VCT) level instrumentation and the fact that the power supply for valves CH-501 and CH-536 was lost due to the shedding of the Motor Control Center (MCC), which supplies these two motor operated valves, on a Safety Injection Actuation Signal (SIAS). Subsequent to this event, ANPP reviewed the system design and proposed modifications to the system that would: 1) improve the operator's ability to operate the system from the control room, 2) provide an automatic function to reduce the amount of required operator action, and 3) improve the reliability of the control grade level instrumentation on the VCT. The result of this review was the implementation of modifications in PVNGS Units 1 and 2. The modifications will be implemented in PVNGS Unit 3 prior to fuel load. The specific modifications are listed below:

1) Provided power to valves CH-501 and CH-536 from a Class 1E MCC that does not get stripped from the bus following a SIAS.

2) Upgraded the VCT level instrumentation to improve the reliability. The new design involves the addition of a new dry reference leg along with the existing wet reference leg. This upgraded level instrumentation also has a signal comparator which provides an alarm to the control room operators in the event of a deviation between the two level channels. This will alert the operators of a potential problem with the VCT level instrumentation.

3) Provided for automatic re-alignment of valves CH-501 and CH-536 on Lo-Lo VCT level and loss of offsite power. This modification allows for the automatic re-alignment of the charging pump suction to the Refueling Water Tank (RWT) gravity feed flow path.

4) Locked open valves CH-532 and CH-524 to ensure the availability of the charging flowpath to the Reactor Coolant System (RCS) and the Auxiliary Pressurizer Spray System (APSS).

Another NRC Staff concern following the September 12, 1985 event was the fact that hydrogen gas had to be vented from the charging pump piping in order to restore the charging pumps. The NRC Staff took the position that this venting of hydrogen to the charging pump cubicles is a hazard

due to the possibility of a subsequent hydrogen burn in these areas that are not ventilated by an essential system. Due to this NRC position, ANPP was required to modify the charging system by installing vent piping which allows the charging pump piping to be vented to an area of the auxiliary building which does receive essential ventilation. The NRC Staff determined that the vent piping modification is acceptable as an interim solution pending the completion of the engineering evaluation which considers alternative hardware modifications to eliminate the need to vent hydrogen from the charging pump piping. This engineering evaluation has been completed and was submitted to the NRC Staff by letter dated June 26, 1986 (ANPP-37162). The following modifications were recommended by the engineering evaluation:

1) Provide emergency backed power to the existing Boric Acid Makeup Pumps (BAMP's).

2) Provide emergency backed power to the BAMP outlet isolation valve to the charging pump suction header (valve CH-514).

3) Provide a new redundant VCT outlet isolation valve in series with the existing VCT outlet isolation valve (valve CH-501). This new valve will also receive emergency backed power.

The ANPP implementation of the above recommended modifications is contingent upon the NRC resolution of Unresolved Safety Issue (USI) A-45 concerning decay heat removal. There is a possibility that the NRC resolution of USI A-45 could result in the backfitting of PVNGS with modifications that would remove the necessity of relying upon the charging system for Branch Technical Position (BTP) RSB 5-1 compliance. It should be noted that the NRC Staff has not yet formally responded to this ANPP proposal. (Refer to the attached figures for simplified diagrams of the charging system.)

Additional charging pump gas binding events have occurred. These events are listed below along with the cause of the event:

1) February 18, 1986 - PVNGS Unit 1 temporarily lost all charging flow due to a failed discharge pulsation dampener.

2) July 12, 1986 - One charging pump in PVNGS Unit 1 became gas bound due to the accumulation of gas in an un-vented suction dampener and a faulty crimp in a wire in the control circuitry for valve CH-536 which prevented the automatic opening of the valve.

3) July 18, 1986 - One charging pump in PVNGS Unit 2 became gas bound due to a failed discharge pulsation dampener bladder.

The following ANPP corrective actions have been implemented in order to prevent the recurrence of the gas binding events:

1) Procedural changes have been implemented to close the charging pump suction valve during a precharge operation to prevent migration of possible leaking nitrogen gas to the other charging pumps. Additionally, the bladder integrity is verified following a precharge by performing a pressure check of the discharge piping.

2) The pulsation dampener bladders are replaced on a refueling outage interval (18-months) versus the manufacturer's recommended three year cycle.

3) A preventive maintenance program has been implemented to periodically vent the process side of the suction stabalizers to remove accumulated gases.

ANPP believes that the implementation of the previous hardware modifications, procedural changes, and maintenance enhancements will significantly increase the availability of the charging system and further ensure that it is capable of fulfilling the auxiliary pressurizer spray function.

7. CONCERN

Masonry walls at the 74 foot elevation of the Control Building.

ANPP RESPONSE

A minimal number of walls at ANPP were designed and installed as masonry fire barriers. The walls were designed as non-load bearing walls (i.e., not part of the building structural system). Design calculations were performed to Seismic Category I criteria (OBE and SSE) and methodology. The drawings and specifications were designated Quality Class S and defined all the requirements to support the design assumptions (including the UBC special inspection). The walls were designated as Seismic Category IX since they did not support but were adjacent to safety related components. Later, small safety related attachments were installed and individually evaluated.

As a result of an NRC CAT inspection of Unit 3, a deficiency in the construction of the Unit 3 and Units 1 and 2 masonry walls was identified. Extensive engineering evaluation of the masonry walls has been performed since, resulting in three presentations and numerous submittals.

On October 14, 1986, ANPP received the NRC Staff's response to the June 19, 1986, submittal. The NRC Staff's response stated that:

(1) The 20 wall panels at elevation 100'-0" of the Control Building are adequate as constructed.

(2) The 3 wall panels at elevation 74'-0" require strengthening to meet industry standards.

(3) Preliminary assessment of the ANPP information submitted on September 19, 1986, leads the NRC to believe that the NRC Staff conclusions will remain unchanged.

As a result of this NRC conclusion ANPP provided a proposed modification to the walls on October 31, 1986.

The modification consists of a series of steel plate assemblies bolted, in pairs, to the masonry walls to "sandwich" them. The design provides both a strengthening effect and a stiffening effect to respond to the concerns expressed in the NRC Staff's initial evaluation.

Since immediate implementation of the masonry wall enhancements is not a significant safety issue, and a short delay would pose no additional seismic risk impact, ANPP proposed that any physical work to enhance the elevation 74'-0" masonry walls on the licensed units would be best achieved during planned plant shutdown. ANPP also stated that it is more appropriate to implement this modification while the nuclear units are not operating based on the facts that the modification requires aggressive construction activities. ANPP proposed the following implementation schedules:

Unit 1 - Complete implementation by first appropriate planned outage of sufficient duration, but not later than first refueling.

Unit 2 - Complete implementation by first appropriate planned outage of sufficient duration, but not later than first refueling.

Unit 3 - Complete implementation by initial criticality.

The NRC has formally responded to the ANPP proposal by letter dated December 19, 1986. The NRC response concluded that the proposed modifications are acceptable subject to three conditions. These conditions have been reviewed and found to be acceptable by ANPP.

8. <u>CONCERN</u>

During the July 12, 1986, PVNGS Unit 1 trip, the control room supervisor first diagnosed the event as a small break LOCA when in reality it was an excessive steam demand event. During the event, there was also a partial loss of non-1E power which energizes certain main control board indicators needed to complete diagnosis of the event. The NRC Staff has transmitted their concerns related to this event in a letter to ANPP dated September 10, 1986.

<u>ANPP RESPONSE</u>

The Staff's recommendations presented in the September 10, 1986, letter have been addressed and procedure modifications implemented.

An additional question was discussed by G. Knighton in the October 28, 1986, meeting with ANPP. This question pertained to the status of the Reactor Coolant Pump tripping criteria change for the Small Break LOCA Emergency Operating Procedure. ANPP has deleted the requirement for tripping all the remaining reactor coolant pumps prior to entering the Small Break LOCA procedure. The tripping of the reactor coolant pumps now occurs after the Small Break LOCA has been verified in the recovery operating procedure when subcooling is lost. This change has been incorporated into the Emergency Operating Procedure diagnostic flow chart and now provides consistency with the Small Break LOCA procedure and the CE Emergency Procedure Technical Guidelines (CEN-152).

9. <u>CONCERN</u>

The NRC noted that the low pressurizer pressure trip and the safety injection actuation setpoints were the same. The question was raised as to whether it was desirable or not to have them the same.

<u>ANPP RESPONSE</u>

The low pressurizer pressure trip is provided to trip the reactor and to assist the engineered safety features in the event of a decrease in Reactor Coolant System (RCS) inventory or in the event of an increase in heat removal by the secondary system. The low pressurizer pressure trip setpoint is currently required to be set at greater than or equal to 1837 psia in PVNGS Units 1 and 2. When the pressurizer pressure decreases to below this trip setpoint, the following actions are initiated: i) reactor trip, ii) Safety Injection Actuation Signal (SIAS), and iii) Containment Isolation Actuation Signal (CIAS). The current PVNGS safety analyses credit the low pressurizer pressure trip for reactor trip actuations and for SIAS/CIAS actuations.

Operating experience gathered at PVNGS to date indicates that the PVNGS Units 1 and 2 reactors have not tripped on low pressurizer pressure. However, there have been several SIAS/CIAS actuations from the low pressurizer pressure trip. Thus, there have been no concurrent reactor trip and SIAS/CIAS actuations from the low pressurizer pressure trip. This is as expected since the SIAS/CIAS actuations have been caused primarily by the post-reactor trip overcooling of the RCS. The only times that you would expect to experience a concurrent reactor trip and SIAS/CIAS actuation from the low pressurizer pressure trip is during an actual loss of RCS inventory event. During a LOCA accident scenario, it is desirable to initiate a reactor trip and SIAS/CIAS actuation as soon as possible after the event initiation in order to limit the consequences of the accident.

10. <u>CONCERN</u>

ANPP has taken exception to provisions in Regulatory Guide 1.68.2 which would require a test to prove remote shutdown capability in each unit. The NRC has determined that the test should be done in each unit. ANPP will respond to the NRC by November 10, 1986.

<u>ANPP RESPONSE</u>

In the PVNGS FSAR, ANPP took exception to the recommendations of Regulatory Guide 1.68.2 in regards to testing of the remote shutdown capability. ANPP stated that a remote shutdown test would be conducted on the first PVNGS unit only. Subsequent to the completion of the PVNGS Unit 2 power ascension test program, the NRC determined that this approach was not acceptable and that a remote shutdown test would be required for PVNGS Units 2 and 3. The NRC sent this determination along with a backfitting analysis to ANPP by letter dated October 9, 1986. ANPP has already responded to this NRC letter (refer to ANPP-39032 dated November 7, 1986). The ANPP response stated that remote shutdown tests would be performed in PVNGS Units 2 and 3 according to the following schedule:

PVNGS Unit 2 = during shutdown for the survillance testing outage scheduled to begin January 9, 1987.

PVNGS Unit 3 = during the power ascension testing program.

11. CONCERN

Problems were experienced with unwanted actuation of breakers in the switchyard. The breaker operations which caused actuations of the Reactor Protection System were caused by malfunctions in the Plant Multiplex System (PMUX). Switchyard breaker control circuits have been hardwired, bypassing the PMUX for this function.

ANPP RESPONSE

In October, 1985, PMUX malfunctions caused two instances of loss of offsite power events. Failure of backup power sources to maintain remote units on line 'froze' all main control board indications of switchyard breaker status, resulting in difficulty in diagnosing the problem, and delaying restoration of power. Evaluation and root cause assessment of the events determined the trips to be a result of the following conditions and failures: i) the plant was in an abnormal electrical lineup in preparation for performance of the subsynchronous resonance test, ii) a component failure in the PMUX Local Multiplexor Terminal (LMT) power supply, and iii) an improperly set jumper on a circuit board in the LMT. Immediate actions were taken in October, 1985 to bypass the PMUX from the critical 13.8 kV switchyard breaker controls by hardwiring, replacing the failed power supply, and correcting and verifying the circuit board jumpering in the LMT. Further long-term items are being implemented such as:

(1) Hardwiring all 13.8 kV switchyard breaker controls.

(2) Air conditioning the switchyard Remote Multiplexor Terminal (RMT) cabinets to preclude heat related failures of electronics and backup power inverter batteries.

(3) Maintenance responsibility for PMUX has been assigned to a single group.

(4) Adding Uninterruptible Power Supply trouble alarms which will alert maintenance personnel of failures in backup power supply sources.

(5) Implementation of a PMUX Reliability Improvement Program to improve operator confidence in PMUX and to improve the system reliability.

12. CONCERN

Numerous computer failures in the security system have been a problem.

ANPP RESPONSE

Security computer problems which have caused or contributed to computer failures are as follows:

1) Inappropriate command software,

2) Inadequate alarm buffer space for incoming alarms,

3) Sticking command console keys resulting in incorrect entries, and

4) A generally awkward operator/computer interface.

These problems have been or are currently being addressed as follows:

1) Improved software has been provided.

2) The computer alarm buffer space has been increased and spurious alarms have been suppressed.

3) A software editor was added which discriminates between valid entries and invalid entries like those resulting from sticking command console keys.

4) A pending design change package is to provide computer operators with an interface that is easier to use.

5) Additional staff have been dedicated to provide security computer maintenance.

The foregoing actions have resulted in far fewer security computer failures. Completion of the operator/computer interface DCP is expected to further reinforce this trend. Each case of security computer failure was properly compensated for at the time of occurrence. Additionally, each failure was an event of short duration, relative to overall security computer availability.

13. CONCERN

PVNGS has demonstrated vulnerabilities to single failures as demonstrated by charging pump pulsation dampener failures, actuation of SBCS by a computer card, startup transformer current transformer connections, and a loose wire on MOV-536.

ANPP RESPONSE

(1) An intense effort is underway to identify single failure vulnerabilities at PVNGS. A scram reduction program has been developed to identify BOP systems and components in which a single failure could result in a reactor trip. This program consists of the following subprograms:

(a) Identification of mechanical and control system single failures which have the potential for resulting in a reactor trip (e.g. - SBCS, FWCS, EHC).

(b) Identification of evolutions (maintenance actions, surveillance tests) which have the potential for resulting in a reactor trip. The systems operating procedures and surveillance tests will then be reviewed for adequate operator cautions to prevent an inadvertent reactor trip.

(c) Identification of reactor protection system failures which have significant potential for resulting in spurious reactor trips.

The program consists of an initial screening phase, an engineering evaluation phase, a quantification phase, an importance ranking and cost benefit analysis phase, and a documentation and final report phase. The primary product of the program will be identification of single failure scram initiators and recommendations for corrective action. Identification of single failure vulnerabilities for systems identified as important to reactor trip reduction will include:

(a) Categorization of root cause of system failures which have previously resulted in reactor trips.

(b) Evaluation of operations experience for identification of system failures which have occurred which have the potential for resulting in a reactor trip.

(c) Where warranted, (e.g. - FWCS, EHC system) operating experience will be supplemented by detailed engineering review of systems using fault tree and/or FMEA techniques as appropriate.

(d) Consideration of lessons learned from previous experience.

The program will include participants from several ANPP departments such as Operations Engineering, Maintenance, Nuclear Engineering, and Nuclear Safety. The Nuclear Analysis group of Nuclear Engineering will have the overall lead and coordination duties of the project. The initial effort will concentrate on control system single failure scram initiators, and the final report will be completed by December, 1987.

(2) The ANPP Nuclear Engineering - Nuclear Analysis Group is developing a Level 1 PRA for PVNGS. This program will provide a deterministic method for identifying potential safety-related single failure vulnerabilities and undesirable system interactions.

14. CONCERN

Numerous annunciators remain lighted on the main control boards.

ANPP RESPONSE

All invalid annunciators for the operating units at PVNGS have been, and continue to be, identified through regular inspection by operational and maintenance personnel.

The result of these inspections are input to a formatted, computerized listing that has been implemented to track any investigative or corrective actions that are required to resolve each invalid annunication. The information supplied by this listing includes annunciator number, annunciator description, problem description, priority level, corrective action, tracking documentation, and current status of corrective action. This listing is reviewed by management, and is available to unit operating personnel so that they can be cognizant of maintenance actions with regard to invalid annunciations.

ANPP is continuing to correct invalid annunciations based on operational, material, and priority restraints. Unit 3 annunciators are expected to have a greater reliability resulting from the PVNGS design change process and management attention during system turnover acceptance.

15. CONCERN

ANPP should develop more formal and regular communication with other plants, particularly the other CPC plants.

ANPP RESPONSE

Arizona Nuclear Power Project has developed formal communications with other utilities through the Combustion Engineering Owners Group (CEOG), NUMARC, and Nuclear Operations Committee.

In addition, ANPP is currently involved in a Core Protection Calculator (CPC)/Core Operating Limit Supervisory System (COLSS) improvement program that was developed through a cooperative effort of other nuclear power facilities that utilize CPC's. Participating in this effort are ANPP, Arkansas Power & Light (Arkansas Nuclear One-2), Southern California Edison Co. (San Onofre-2/3), Louisiana Power & Light Co. (Waterford 3), Washington Public Power Supply System (WNP-3), and Combustion Engineering Co. Although there is not a formal charter that establishes the guidelines of this cooperation, it is referred to as the CPC Oversight Committee and it has met on an as-needed basis to resolve generic CPC issues. ANPP has taken on active role in this committee and continues to correspond and communicate regularly with the other participants.

16. CONCERN

The NRC has expressed concern about whether or not formal procedures existed to ensure the transfer of information and data from one unit to another.

ANPP RESPONSE

ANPP has existing programs in place which provide for information dissemination throughout PVNGS. The following items are some examples of the programs in place at PVNGS:

(1) Operations Department Experience Report (ODER) -

The ODER is designed specifically to provide a communications vehicle for the Operations Department to disseminate information such as lessons learned from plant trips, operating experience, events at other nuclear power plants, etc. The ODER program is included in the PVNGS Operations Department Guidelines. The ODER program provides a means of disseminating operating information from one unit to the other PVNGS operators.

(2) Interdepartmental Event Investigation -

The Interdepartmental Event Investigation is a newly implemented program designed to provide a consistent methodology for investigating events and identifying necessary corrective actions. This program is specifically oriented to address personnel errors and procedural violations. The results of the investigations are provided to all personnel that might be subject to similar incidents. The procedure which governs the Interdepartmental Event Investigation is 71AC-0ZZ03 in the PVNGS Station Manual. For further details on how this procedure is applicable to addressing personnel errors see the response to Concern #23.

(3) Quality Talks/Safety Speaks -

Procedure 6N417.20.00 of the ANPP Policies and Procedures Manual is the Quality Talks/Safety Speaks program. This program is designed to keep employees informed of current project safety problems, quality problems, and other areas of concern such as work practices, violations, trends, etc.

In addition to the programs described above, some PVNGS Departments are provided with their own support group which aids the department in investigating events, root causes of malfunctions, special problems and dissemination of information to each unit's dedicated staff to ensure consistency between the PVNGS units. Examples of the interdepartmental support groups are: Operations Support, Maintenance Support, Radiation Support, etc.

In the event that difficult and/or complex problems arise in which several departments are involved, a special task force can be formed to address the broad scope issues requiring focused attention from the various departments. This type of task force provides a defined organization to resolve the problem. A recent example of where a task force has been formed to resolve difficult problems is in the area of addressing our earlier security deficiencies.

17. CONCERN

Concern was expressed by the NRC concerning load shedding causing valve inoperability. The NRC wanted to know if ANPP is looking at this problem.

ANPP RESPONSE

PVNGS Unit 1 has experienced two events where load shedding has resulted in valve inoperability. The valve inoperability has resulted in degraded plant conditions. The first event occurred on September 12, 1985, and resulted in a loss of power to valves CH-501 and CH-536 in the charging system. These two valves received power from 480V Motor Control Center (MCC) NHN-M72. This MCC receives power from Class 1E load center PGA-L35. Since MCC NHN-M72 is a non-class MCC, it gets shed from the Class 1E load center on a Safety Injection Actuation Signal (SIAS). During the September 12, 1985 event, a SIAS was received and MCC NHN-M72 was shed which resulted in a loss of power to valves CH-501 and CH-536. The valves failed as-is on loss of power and resulted in gas binding of the charging pumps. As a result of this event and the importance of retaining power to these valves, the power supply to these valves was modified. Valves CH-501 and CH-536 now receive power from a Class 1E MCC that does not get shed from the Class 1E distribution system on a SIAS. Thus, the modification ensures the operability of these two valves following a SIAS which further ensures the operability of the charging system.

The second event occurred on July 12, 1986, at PVNGS Unit 1. The event was initiated by a reactor and turbine trip. The grid perturbation due to the turbine/generator trip was sufficient to cause a load shed of all non-class load centers on 13.8 kV buses NAN-S01 and NAN-S02. This load shedding resulted in a loss of power to the 2nd stage reheater steam source valves. These valves are motor operated valves that fail as-is on a loss of power. In this case, the valves failed in the open position and created four steam paths from the main steam header to the condenser. This significantly contributed to the post-reactor trip cooldown of the primary system. Modifications have been implemented to prevent a similar event from occurring (refer to Concern #4 for a description of these modifications).

In summary, these previous operating events as well as the loss of offsite power tests (conducted during the power ascension testing programs) have provided valuable information on plant response to load shedding events. Corrective measures have been implemented to ensure that these valves operate in a manner to ensure plant safety during a load shed.

18. CONCERN

The NRC noted that we must ensure that all information for the Sholly process is included in licensing submittals. In addition, consideration must be given to changes which could be affected by the Diablo Canyon decision in the Ninth Circuit Court which opened hearings for certain kinds of changes. Change requests should look at more than just the small scope of the change, and the time when the changes are needed must be considered.

ANPP RESPONSE

The concern with the NRC has been the adequacy of ANPP technical specification amendment requests. ANPP has been working closely with the NRC Staff to identify specific guidelines to follow to improve the submittals. A meeting was held in Bethesda, MD on December 9, 1986 to discuss and clarify the process. The NRC Staff is currently reviewing our latest submittal and will provide feedback on it.

In the interim, ANPP is striving to follow NRC guidance on any technical specification amendment submittals, and will continue to communicate regularly with the NRC Staff on this subject.

19. <u>CONCERN</u>

Does the Post Trip Review Procedure adequately cover what must be done and what must be monitored when the plant is restarted, when the cause of the trip has not been fully determined.

<u>ANPP RESPONSE</u>

The Post Trip Review Reporting procedure at ANPP has undergone several enhancements in the area of root cause analysis. The following is a description of the current procedure.

Following a reactor trip event, the duty Shift Technical Advisor (STA) and the Unit Superintendent (or his designee) work together to formulate a preliminary event sequence and description. The Post Trip Evaluation Team (PTET) then meets to perform a detailed event analysis. The PTET is chaired by the Unit Superintendent and its members include supervisory personnel in maintenance, operations, engineering, and technical services. Additional personnel are called in if deemed necessary by the Unit Superintendent. Using a defined list of information, the PTET identifies concerns and allocates resources to perform root cause analyses and suggest corrective actions. These analyses are performed to determine the adequacy of system/component response, operator actions, procedure use, and procedure effectiveness.

Following completion of the investigations and analyses performed by the various working groups of the PTET, this group meets again to evaluate the results and to formulate an action plan which is submitted to the Post Trip Management Review Team (PTMRT). The PTMRT is chaired by the Plant Manager and includes upper plant management personnel. The PTMRT reviews the plan and must concur with it prior to implementation.

When work is authorized to proceed, the PTET is kept apprised of progress. When the root cause is identified and corrected, or all possible troubleshooting avenues have been exhausted for each safety significant identified concern, the PTET recommends approval of the Post Trip Review Report (PTRR) by the PTMRT.

The PTMRT reviews concerns, actions, and resolutions and approves initial mode 2 entry according to the following criteria:

- Root cause identified and corrected, or
- All possible troubleshooting avenues have been exhausted, and
- Any open items are justified with bases as to why restart is satisfactory in spite of the open items.
- If the open items constitute a change from normal plant configuration or design, a 10CFR50.59 review and evaluation is performed.

When all concerns are addressed and the restart criteria has been met, restart approval is given by the PTMRT, following a technical review by an STA or licensed SRO individual and approval recommendation of the PTET.

20. CONCERN

Does the Safety Parameter Display System (SPDS) computer have sufficient capacity to perform in a timely manner if a third unit is added to the system?

ANPP RESPONSE

Since the submittal of the schedular extension request for the SPDS dated April 22, 1986 (ANPP-36303), ANPP has made several software changes to the SPDS to increase the system's availability and capacity. These modifications were discussed with the NRC Staff during their PVNGS SPDS audit held November 18 - 20, 1986 at PVNGS. It should be noted that during this audit the SPDS was receiving information from Unit 3 as the unit was in the hot functional testing phase. Therefore, we do not expect to encounter any problems with the addition of PVNGS Unit 3 to the SPDS.

21. <u>CONCERN</u>

There have been control problems at low power levels in the feedwater control system.

<u>ANPP RESPONSE</u>

The Feedwater Control System (FWCS) at PVNGS is currently being evaluated, by on-site and off-site engineering departments, for revision to the feedwater pump speed control program setpoints. This evaluation resulted from observations that attempts to place the feedwater pump speed controller in automatic above 15% power caused oscillations in the FWCS master controller. These oscillations appear to be caused by a feedwater pump speed minimum setpoint that is too high and causes an excessive pressure drop/flow change across the feedwater control valves at low power levels.

The engineering personnel at ANPP are in the process of developing a simulation model of the FWCS to determine optimal setpoints. In the interim, it has been at the discretion of the shift supervisor whether to institute automatic control of feedwater pump speed at low power and closely monitor oscillations, or to remain in manual until a higher power level is reached when the FWCS will be more stable.

22. CONCERN

PVNGS has had several condenser tube problems. PVNGS has titanium tubes and relatively little debris to enter the water boxes. Condenser tube performance should be better.

ANPP RESPONSE

PVNGS has experienced several condenser tube failures and has implemented appropriate corrective actions to minimize future tube failures and to improve plant availability. The condenser tube failures are attributed to the following conditions:

1) High cycle fatigue failures of condenser tubes have been caused by high velocity steam flow through the condenser and the sequencing of the steam bypass valves into the condenser. The original steam bypass valve sequencing was such that at low bypass demand conditions where only a couple of valves are required, the first three selected valves would all dump to the "A" condenser shell. To correct these causes of high cycle fatigue tube failures, the condenser vacuum pressure was raised to lower the steam velocity and the steam bypass valves were re-sequenced and baffled to redirect the steam flow. The re-sequencing of the steam bypass valves equalizes the loading on the condenser shells by selecting the valves such that the first three selected valves are each directed to a different condenser shell. Additionally, tube stakes have been installed in the lowest pressure shell ("A" shell) to reduce tube vibrations.

2) Tube failures have been experienced due to impingement from steam or water dumps. The individual impingement sources have been corrected as they were identified. Additionally, the operators have been made aware of the proper manner of condenser dump operation in order to avoid operation outside of the design conditions of the condenser.

3) A number of the condenser tube problems have been attributed to power ascension testing. Firstly, the power ascension testing phase is the first time that the condenser is subjected to normal operating loads. Thus, tube failures are most likely to occur at this time. Secondly, the condenser is operated at partial loads for extended periods of time during the power ascension testing phase. Since the condenser is primarily designed for full power operation, some tube failures are attributable to this extended operation at partial loading conditions.

23. CONCERN

PVNGS has had several personnel errors/technical specification violations. Describe actions taken by management to reduce these occurrences.

ANPP RESPONSE

The ANPP Maintenance I&C department implemented a Quality Improvement Report (QIR) program in September 1985. The functions of the QIR program are to document and communicate improvements/corrective actions to prevent occurrence or recurrence of undesirable events/activities. As a result of the success of the QIR program, its concept has been incorporated into the Interdepartmental Event Investigation (IEI) program which is implemented throughout PVNGS.

Plant management instituted the IEI program (71AC-0ZZ03) in October, 1986. Under this program, all personnel errors are investigated by the affected department manager. The findings are reported through the manager's immediate supervisor to the PVNGS Plant Manager. The Plant Manager determines if an interdepartmental board should be convened. The board consists of the Plant Manager and selected direct reports, and recommends what actions, including disciplinary, should be taken. The ANPP Compliance Department is responsible for disseminating the results of the investigations to other departments, and for retention and trending of the investigative reports.

ANPP believes that these programs will significantly reduce the number of personnel errors/technical specification violations by ensuring that the causes of the events are determined and that appropriate actions are taken to prevent recurrence.