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FACIL: STN-50-528 Palo Verde Nuclear Station, Unit 1, Arizona Publi 05000528
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RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 88-016-00: on 880514, reactor trip occurred. Caused by non-conservative operator performance during reactor startup. Appropriate procedure precautions implemented to ensure awareness from control room personnnel. W/880613 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 12
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: Standardized plant. 05000528

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	NRR/DLPQ/QAB 10	1	1	NRR/DOEA/EAB 11	1	1
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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Palo Verde Unit 1	0 5 0 0 0 5 2 8	1 OF 1 1

TITLE (4)
Reactor Trip Following Earlier Than Anticipated Criticality

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
									N/A	0 5 0 0 0
0 5	1 4	8 8	8 8	– 0 1 6	– 0 0	0 6	1 3	8 8	N/A	0 5 0 0 0

OPERATING MODE (9)	3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)		20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Timothy D. Shriver, Compliance Manager	AREA CODE 6 0 2 — 3 9 3 – 2 5 2 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 8	1 5	8 8

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At approximately 0335 MST on May 14, 1988 Palo Verde Unit 1 was in Mode 3 (HOT STANDBY) when a reactor trip occurred as the Control Element Assemblies (CEA's) (AA) were being inserted following an attempt to startup the reactor. The trip occurred when overly conservative Radial Peaking Factors (RPF) were utilized by the Core Protection Calculator (CPC) (CPU) (JC) as the CEA's were being inserted. There were no other safety system responses (including ESF actuations) and none were necessary. The plant was immediately stabilized in Mode 3.

The CEA's were being inserted after criticality had been achieved earlier than calculated resulting in the CEA's being below the Power Dependent Insertion Limits of LCO 3.1.3.6. The root cause of the criticality outside established guidelines has been determined to be non-conservative operator performance during the reactor startup. Errors in the information utilized for calculating the Estimated Critical Condition (ECC) contributed to this event.

The corrective action to prevent recurrence will be to correct the errors in the information utilized for the ECC and improve the administrative controls for utilizing the ECC. Appropriate disciplinary action will be taken.

There have been no previous similar events reported pursuant to 10CFR50.73; however, a reactor trip did occur as a result of overly conservative RPF's being utilized by the CPC as reported in Unit 1 LER 88-011-00.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Palo Verde Unit 1	0 5 0 0 0 5 2 8	8 8	— 0 1 6	— 0 0	0 2 OF 1 1

TEXT *(If more space is required, use additional NRC Form 366A's)* (17)

I. DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

On May 14, 1988, Palo Verde Unit 1 was in Mode 3 (HOT STANDBY) at normal operating temperature and pressure. A reactor startup was in progress following a trip from 91 percent power which had occurred approximately 38.5 hours earlier.

B. Reportable Event Description (Including Dates and Approximate Times of Major Occurrences):

Event Classification:

Automatic actuation of the Reactor Protection System. Condition prohibited by the plant's Technical Specifications.

On May 14, 1988, Palo Verde Unit 1 was in Mode 3 (HOT STANDBY) conducting a reactor (AC) (RC) startup. During the reactor startup, the reactor achieved criticality prior to that calculated by the Estimated Critical Condition (ECC). As criticality was achieved below the Power Dependent Insertion Limits of Limiting Condition for Operation 3.1.3.6, it was decided to insert Control Element Assemblies (CEA)(AA) to calculate a new ECC. As the CEA's were being inserted, a reactor trip occurred at approximately 0335 MST on May 14, 1988. The reactor had been shutdown for approximately 38.5 hours prior to the trip. The Estimated Critical Rod Position per the ECC was 90" withdrawn on Regulating (Reg) Group 4 with a boron concentration of 1033 ppm for a startup time of 0000 MST.

The startup began at approximately 0100 MST by withdrawing the Shutdown (SD) CEA's banks and the Part Length CEA's (PLCEAs). The operating crew (utility, licensed) completed withdrawal of the SD banks and the PLCEA's at approximately 0159 MST. Withdrawal of the Regulating Groups began at approximately 0304 MST.

The count rate, obtained from the Startup Channels (IG) (XI), was approximately 300 counts per second (cps) when Reg Group 1 was 0 inches withdrawn. The startup was conducted in accordance with 41OP-1ZZ03, "Reactor Startup", with the regulating CEA's being withdrawn in 30 inch increments per step 4.3.12. After each withdrawal increment, a pause was established to allow count rate/power level to stabilize. Additionally, the Shift Technical Advisor (STA) (utility, licensed) was recording count rate after each 30 inch withdrawal. This was started when Reg Group 1 was being withdrawn even though the procedure only requires that power level be recorded and plotted with each 30 inch withdrawal after reaching 60 inches withdrawn on Reg Group 3 and thereafter.

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Palo Verde Unit 1	0 5 0 0 0 5 2 8	8 8	— 0 1 6	— 0 0	0 3 OF 1 1

TEXT (If more space is required, use additional NRC Form 366A's) (17)

The Primary Operator (utility, licensed) complied with section 4.3.12 of the procedure and withdrew Reg Groups 1 and 2 in 30 inch increments. When Reg Group 3 was withdrawn to 30 inches, the Primary Operator (utility, licensed) questioned the STA concerning count rate and was told that it had stabilized (the STA noted that the count rate was approximately 1277 cps). Count rate was noted to have doubled twice since beginning the withdrawal of Reg Group CEA's. Since criticality was imminent, the Control Room Supervisor (CRS) (utility, licensed) checked the Power Dependent Insertion Limits (PDILs) of Specification 3.1.3.6. Technical Specification LCO 3.1.3.6 specified that in order to enter Mode 2 (STARTUP), the CEAs in Reg Group 3 must be at least 60 inches withdrawn. With the count rate stable at approximately 1277 cps, the Primary Operator pulled Reg Group 3 to 45 inches withdrawn. While the CEA's were being withdrawn to 45 inches, the startup channels (IG) were deenergized in accordance with the procedure at approximately 2000cps. Power level was then monitored on the log power channels (IG) after observing proper overlap on the startup channel and log power channel.

Upon reaching 45 inches withdrawn on Reg Group 3, the startup rate was still not definitely positive and power level had stabilized. The Primary Operator therefore commenced pulling Reg Group 3 to 60 inches withdrawn. The CEA withdrawal was made in three distinct steps taking between 1 and 5 minutes to complete. After the 15 inch withdrawal, the CRS concluded that the reactor was slightly supercritical and, hence, the critical CEA position was between 45 inches and 60 inches. (Note: The measure of criticality is actually based on the indication of a positive startup rate and an increasing power level without CEA motion. Thus, the reactor is actually brought to a supercritical condition.)

The CRS directed the Primary Operator not to allow power to exceed 1E-03 percent. The Primary Operator initiated CEA insertions to stabilize power at less than 1E-03 percent power. The CRS then conferred with the Shift Supervisor on what action to take. They concurred that it would be inappropriate to be critical while not meeting the PDIL requirements. They decided to insert Reg Group 3 to 0 inches withdrawn and investigate the deviations from the ECC. The direction to insert Reg Group 3 to 0 inches was given to the Primary Operator who then complied. It should be noted that Reg Group 3 was 60 inches withdrawn for approximately 2 minutes, 39 seconds.

When the CEA's reached approximately 25 inches withdrawn, an auxiliary trip was generated by Core Protection Calculators (CPC) (CPU) (JC) Channels B and C on high Radial Peaking Factors. The Reactor Trip Switchgear (SWGR) operated as designed, and CPC channels "A" and "D" tripped as expected.

The plant was immediately stabilized in Mode 3. The event was diagnosed by the Assistant Shift Supervisor (utility, licensed) as an uncomplicated Reactor trip and performance of the appropriate procedure was initiated.

The following information concerns the investigation into the cause of the trip.

The CPC trip buffers are not reset until the critical rod height data is taken, as stated in ANPP procedures. The CPC's cannot be reset unless Reg Group 3 is withdrawn sufficiently to reduce the integrated one-pin peak below the auxiliary trip setpoint (at this time, that position was approximately 27 inches withdrawn). Additionally, 41OP-1ZZ03 calls for the CPC reset when Group 3 is 97 inches withdrawn; this accounts for possibly higher peaks at other conditions. This resulted in a loss of actual trip data from the CPC's which would have verified the presence of the auxiliary trip. Using the CPC Simulator, it was later verified that at less than 30 inches withdrawn on Reg Group 3, an auxiliary trip was correctly generated by the CPC's due to high Radial Peaking Factors. Even though the actual trip buffers for the event were unavailable, the re-creation of the event using the CPC Simulator verified that this was the cause of the reactor trip.

The reactor was subcritical at the time of the trip. No Engineered Safety Features (ESF) actuations were received or required. The Emergency Plan was not initiated and no emergency classification was made.

During ANPP's Post Trip Review evaluation, it was determined that the reactor had gone critical between 50 and 55 inches withdrawn on Group 3. Based upon criticality being achieved below 60 inches withdrawn, Unit 1 operated in a condition prohibited by Technical Specification 3.0.4 in that Mode 2 (STARTUP) was entered without meeting the conditions of LCO 3.1.3.6.

C. Status of structures, systems or components that were inoperable at the start of the event that contributed to the event:

Not applicable - no structures, systems, or components were inoperable at the start of the event which contributed to the event.

D. Cause of each component or system failure, if known:

Not applicable - no component or system failures occurred.

E. Failure mode, mechanism, and effect of each failed component, if known:

Not applicable - no component failures occurred.

F. For failures of components with multiple functions, list of systems or secondary functions that were also affected:

Not applicable - no component failures occurred.

G. For failure that rendered a train of a safety system inoperable, estimated elapsed time from the discovery of the failure until the train was returned to service:

Not applicable - no failures occurred which rendered a train of a safety system inoperable.

H. Method of discovery of each component or system failure or procedural error:

There were no component or system failures involved. The errors discussed in Section I below were identified during the post trip review process conducted by ANPP.

I. Cause of event:

The cause of the reactor trip was an Auxiliary Trip generated by the CPC's. The Auxiliary Trip resulted from conservatively high Radial Peaking Factors being generated as Regulating Group 3 CEA's were being inserted below 30 inches. In general, the conservatively high Radial Peaking Factors may result in a reactor trip when Group 3 is less than 95 inches withdrawn and the CPC's are not bypassed.

The cause of the condition prohibited by the plant's Technical Specifications wherein the reactor achieved criticality below the limits of LCO 3.1.3.6 has been determined to be operator performance which was considered to be less conservative than appropriate for the situation during the reactor startup. It was determined that the control room personnel (utility, licensed) did not act with the desired conservatism in performing the approach to criticality based upon the information available at the time. During the approach to criticality, the control room personnel correctly performed and followed procedures and responded to alarms and permissives to bypass High Log Power trips. However, ANPP Management considers that the degree of conservatism utilized based upon indications of early criticality were not in accordance with management expectations and are considered to be cognitive personnel errors on the part of control room supervision (utility, licensed). As a result of this concern, ANPP performed a Control Room Staff Evaluation. The results of this evaluation are provided in Section V. There were no unusual

characteristics of the work location which contributed to this event.

Contributing to the non-conservatism exhibited by the control room personnel, some of the information being utilized by the control room personnel was determined to be incorrect and/or inadequate. The ECC being utilized by control room personnel contained inaccuracies which resulted from: (1) an inaccuracy in the computer program which calculates transient xenon level and (2) a startup procedure which allowed a 4 hour deviation from the projected startup time (At the time of the approach to criticality, approximately 3.5 hours had elapsed from the projected startup time. During this time period Xenon decay caused a positive reactivity change). The boronometer (XI) being utilized for determining boron levels in the Reactor Coolant System (RCS) (AB) may not have provided accurate indication of boron concentration (This issue is being evaluated by engineering and appears to be due to a non-linear response to variations in boron concentration). The information and controls available for use by control room personnel in evaluating the conditions present during the approach to criticality were determined to be inadequate. That is, based upon the fact that the Core Data Book did not contain integrated CEA worth curves for Group 3 below 60 inches, an inverse count ratio plot (1/M plot) was not required by procedure to be started until Group 3 reached 60 inches withdrawn.

J. Safety System Response:

Reactor Protection System Actuation occurred at approximately 0335 MST on May 14, 1988.

There were no other safety system responses (including ESF actuations) and none were necessary.

K. Failed Component Information:

Not applicable - there were no failed components.

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT:

There were no safety consequences or implications resulting from this event. As described above, the reactor tripped as designed and all safety responses necessary to place the plant in a stable condition functioned properly.

The criticality earlier than calculated in the ECC had no adverse safety consequences or implications. As described above, Unit 1 entered Mode 2 with the CEA's below the transient PDIL limit of Specification 3.1.3.6. Operation in this condition is permitted for up to two (2) hours pursuant to ACTION "a" of LCO 3.1.3.6. The CEA's were below the PDIL limit for

less than 10 minutes. It should be noted that the PDIL limits of Specification 3.1.3.6 are established to ensure that an adequate shutdown margin is maintained and at the same time ensure that the potential effects of a CEA ejection accident are limited to acceptable levels. The function of the shutdown margin requirements is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. Shutdown margin requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature. The most restrictive condition occurs at the end of core life, with cold leg temperature at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified shutdown margin is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. An analysis of the conditions present during the event has determined that the boron concentration was approximately 120 parts per million greater than necessary to meet shutdown margin requirements.

III. CORRECTIVE ACTIONS:

A. Immediate:

When control room personnel (utility, licensed) noted that criticality had been achieved earlier than calculated in the ECC, appropriate actions were taken to shutdown the reactor and place it in a safe condition by inserting Group 3 to zero inches until the problems with the ECC could be investigated.

As described above, the reactor trip occurred as the CEA's were being inserted below approximately 25 inches withdrawn. Following the trip, control room personnel (utility, licensed) took the appropriate action to ensure that the plant was in a safe condition.

B. Action to Prevent Recurrence:

Appropriate procedure precautions have been implemented to ensure that control room personnel are aware that reactor trips may occur if Regulating Group 3 CEA's are less than 95 inches withdrawn and the CPC's are not bypassed.

Concerning the cognitive personnel errors described in Section I.I wherein non-conservative operator performance was involved, appropriate disciplinary action and/or counseling will be taken.

Concerning the error in the ECC, the following actions are being taken:

- Additional controls concerning the time allowance between the time the ECC is calculated and the actual approach to criticality will be developed.

- The computer program which calculates transient xenon levels has been modified.

- RCS boron samples will be utilized for plant startup in lieu of boronometer readings until the instrumentation is verified to be accurate for all plant conditions.

- Information and direction for starting inverse count ratio plots earlier in the startup process will be developed.

- An engineering analysis on the existing ECC calculation methodology will be performed. Based upon this analysis, appropriate controls or changes will be delineated.

Concerning the information and methodology for starting up the reactor, the following corrective actions are being taken:

- The integrated CEA worth curves below 60 inches have been included in the Core Data Book.

- The reactor startup procedure will be revised as appropriate to include the information contained in the Core Data Book.

- A reactor engineer (utility, non-licensed) will be required to be in the control room (NA) during reactor startups until the appropriate administrative changes are made.

As a result of the Control Room Staff Evaluation, the following corrective actions will be taken:

- a review of the Control Room communications during this event will be conducted and guidance on declaring criticality will be promulgated.

- management will issue a letter reminding all plant personnel to adopt a conservative approach when conditions or indications are other than expected.

- a Human Performance Evaluation System evaluation will be performed by the STA Group.

IV. PREVIOUS SIMILAR EVENTS:

There have been no previous similar events reported pursuant to 10CFR50.73 involving a reactor trip following a criticality earlier than anticipated by the ECC. However, a similar trip occurred as reported in

Unit 1 LER 88-011-00 when overly conservative Radial Peaking Factors (RPF) utilized by the CPC resulted in a reactor trip. As discussed in LER 88-011-00, the conservative RPF values are part of the original design of the CPC software. ANPP is currently evaluating the feasibility of modifying the existing software.

V. ADDITIONAL INFORMATION

A. The following information was developed as a result of a Control Room Staff evaluation conducted by ANPP:

SHIFT SUPERVISOR (Utility, Licensed)

The Shift Supervisor (SS) was in the "horseshoe" area. It was his intention to maintain a broad perspective on overall plant response and therefore was not directly involved with the specifics of the criticality. When he was consulted about the PDILs and the critical rod position by the CRS, he concurred with the CRS's recommendation that the Group 3 CEA's be reinserted to 0 inches. ANPP believes the Shift Supervisor should have been more involved in this evolution.

CONTROL ROOM SUPERVISOR/ASSISTANT SHIFT SUPERVISOR (Utility, Licensed)

The CRS was directing the Reactor Startup activities. The CRS was using the correct procedure for the evolution. The Startup was proceeding in a controlled and "unhurried" manner. The CRS had discussed the potential for an "early" criticality due to Xenon decay with his Reactor Operators. The Primary Operator indicated he understood the discussion.

When Group 3 was at 30 inches, it was apparent that, based on the count rate information, the reactor would go critical "...very close to 60 inches...". Due to the apparent large difference between the suspected early criticality of approximately 60 inches on Group 3 and the ECC of 90 inches" on Group 4, the CRS should have taken a more conservative approach and reevaluated the ECC prior to continuing the startup. When Reg Group 3 was at 60 inches, the CRS recognized that the reactor had gone critical during the last rod withdrawal. He then directed the Primary Operator to maintain reactor power less than 1E-03 percent of rated thermal power while he consulted with the SS. At this time, the CRS was primarily concerned with Technical Specification limits on CEA position (PDILs).

It was the understanding of the CRS that the Reactor Operator actually pulling CEAs is the one who actually "calls" criticality. The CRS, upon recognizing that the reactor was critical, asked the Primary Operator, "What are the indications of criticality?". This was done in order to prompt the operator to "call" criticality. In

this case the CRS should have been more direct with his communications to the Primary Operator with regard to what information he wanted with respect to the condition of the reactor, i.e., by asking "Is the reactor critical?". It should also be recognized that there are no formal guidelines regarding who on the Control Room staff should or must "declare criticality." ANPP Management believes that the CRS should have directed the evolution be stopped when it became apparent that the criticality could be achieved earlier than anticipated.

Following the Reactor Trip, the CRS directed the Operators to maintain their safety functions and the plant was stabilized in Mode 3.

NO III - PRIMARY OPERATOR (Utility, Licensed)

The Primary Operator was pulling the CEA's under the direction of the CRS. He observed the power level increase above the point where the Log Power Channel could be bypassed and the CPC channels become "active". Based on the interview with the Primary Operator, he believed the reactor to be critical at approximately 60 inches withdrawn on Reg Group 3. Actions were taken by the Primary Operator to insert the CEA's in order to maintain the reactor at less than 1E-03 percent power at the direction of the CRS. Before the reactor was stabilized and the critical point data could be taken, it was decided to reinsert Group 3. Therefore, criticality was not formally stated nor entered in the Control Room logs. Criticality should have been entered in the Control Room logs as a late entry.

The indications present with Group 3 at 30 inches indicated that subsequent withdrawals would be very near, if not at, criticality. The Primary Operator should have shown more concern with these indications, and at least questioned, the CRS. A more conservative action would have been to recalculate the ECC prior to continuing the Startup. The Primary Operator should have recognized indications of criticality prior to being "prompted" by the CRS.

ANPP believes the Primary Operator should have stopped the evolution when it became apparent that criticality would be achieved earlier than anticipated.

NO III - SECONDARY OPERATOR (Utility, Licensed)

The Secondary Operator was performing the Main Turbine Warmup in preparation for secondary plant startup.

NO III - CONTROL ROOM (Utility, Licensed)

Was not directly involved in startup.

SHIFT TECHNICAL ADVISOR (Utility, Licensed)

The Shift Technical Advisor (STA) was observing the progress of the startup and recorded count-rates periodically during withdrawal of the Shutdown groups and Regulating groups. He indicated that the count rates had doubled twice during the course of the rod withdrawal. The STA should have been more aggressive in providing this information to the Control Room staff. This would have provided additional indication to the Control Room on their nearness to criticality. Since the CEA worth curves are not available in the Core Data Book, it was not possible to perform a 1/M plot.

ANPP Management believes that the STA should have been more involved in monitoring the startup activities and providing direct communication that the reactor was nearing criticality. He should have recommended to the SS that the evolution be stopped when it became apparent that criticality could be achieved earlier than anticipated.

B. Following the event, it was determined that the information provided in the 4-hour call made via the Emergency Notification System (ENS) was not accurate. During the ENS notification, it was discussed that the reactor trip occurred as the CEA's were being inserted in order to calculate a new ECC, and the CEA's were being inserted since the reactor was approaching criticality prior to the ECC. However, it was not discussed that the reactor had achieved earlier criticality and the CEA's were also being inserted due to concerns about meeting PDIL limitations.

ANPP believes that the criticality and PDIL concerns should have been discussed in the initial report.

Investigation into this aspect of the event is continuing and will address whether additional reporting requirements were applicable. Based upon the results of the investigation, corrective actions will be implemented as appropriate. However, as an immediate corrective action additional administrative controls will be implemented to provide more explicit directions for NRC notifications.

C. Exact discussions of the event were impacted by information available in the various logs. ANPP will evaluate this aspect and determine if changes are required to enhance the current log keeping techniques.

D. As previously discussed, additional evaluations/investigations are being conducted as a result of this event in both the reporting/notification aspects and in the area of Human Performance Evaluation System. Based upon the results of these evaluations a supplement to this report will be issued.

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

192-00383-JGH/TDS/DAJ
June 13, 1988

U. S. Nuclear Regulatory Commission
NRC Document Control Desk
Washington, D.C. 20555

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 1
Docket No. STN 50-528 (License No. NPF-41)
Licensee Event Report 88-016-00
File: 88-020-404

Attached please find Licensee Event Report (LER) No. 88-016-00 prepared and submitted pursuant to 10CFR 50.73. In accordance with 10CFR 50.73(d), we are herewith forwarding a copy of the LER to the Regional Administrator of the Region V office.

If you have any questions, please contact T. D. Shriver, Compliance Manager at (602) 393-2521.

Very truly yours,

J. G. Haynes
Vice President
Nuclear Production

JGH/TDS/DAJ/kj

Attachment

cc: E. E. Van Brunt, Jr. (all w/a)
J. B. Martin
T. J. Polich
E. A. Licitra
A. C. Gehr
INPO Records Center

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