APS

10CFR50.73(a)(2)(i)(B)

Palo Verde Nuclear
Generating Station

William E. Ide
Vice President
Nuclear Production

TEL (623) 393-6116
FAX (623) 393-6077

Mail Station 7602
P.O. Box 52034
Phoenix, AZ 85072-2034

192-01067-WEI/AKK/DGM/RJH
May 18, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 50-530
Licensee Event Report 98-006-00

Attached please find Licensee Event Report (LER) 50-530/98-006-00 that has been prepared and submitted pursuant to 10CFR50.73. This LER reports a condition where a degraded Steam Generator tube (wear induced) was not removed from service by plugging in accordance with the Technical Specification required action of the Steam Generator inspection program. A review of Steam Generator inspection results indicated there is firm evidence that the degraded tube condition existed since the previous Unit 3 outage in October of 1998. This condition was discovered on April 18, 2000 during the eighth Unit 3 refueling outage.

No commitments are made to the NRC in this submittal.

In accordance with 10CFR50.73(d), a copy of this LER is being forwarded to the Regional Administrator, NRC Region IV and to the Resident Inspector. If you have questions regarding this submittal, please contact Daniel G. Marks, Section Leader, Regulatory Affairs, at (623) 393-6492.

Sincerely,

Will E Ide

WEI/AKK/DGM/RJH/kg
Attachment

cc: E. W. Merschoff (all with attachment)
J. H. Moorman
M. B. Fields
INPO Records Center

IE22

BGH-001

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Palo Verde Nuclear Generating Station-Unit 3	05000530	1 OF 6

TITLE (4)

Technical Specification for Steam Generator Tube Inspection not Met

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 NAME	DOCKET NUMBER
10	15	1998	1998 -	006 -	00	05	18	2000	N/A	
									FACILITY NAME: N/A	DOCKET NUMBER

OPERATING MODE (9): 6

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Daniel G. Marks, Section Leader, Nuclear Regulatory Affairs	623-393-6492

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 18, 2000, at approximately 1600 mountain standard time (MST), Unit 3 was in MODE 6, refueling, when engineering discovered a wear indication in Steam Generator 3-2. The tube at Row 31, Line 110 in Steam Generator number (SG) 3-2 was plugged due to a wear-induced indication that exceeded the 40% plugging limit of Technical Specification (TS) 5.5.9.4.a.6. Subsequent review of historical data revealed that this particular wear indication was present, but not identified, during the previous Unit 3 refueling outage (U3R7) in October, 1998. Re-evaluation of the October 1998 SG inspection data estimates that the wear indication at that time was approximately 45% through wall. Palo Verde Steam Generator Projects personnel identified that SG 3-2 was returned to service during the following operating cycle (October 1998 to April 2000), with the tube at Row 31, Line 110 having wear greater than 40% through wall, resulting in a condition prohibited by Technical Specifications. Evaluation of this flaw concluded that there is no safety significance for this event in that structural integrity is conservatively assured. Immediate corrective actions were taken to stake and plug the tube. The cause of the missed indication was an isolated failure of the data analyst to identify the tube defect.

I. REPORTING REQUIREMENT(S):

This LER (50-530/98-006-00) is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B) to report an event that resulted in a condition prohibited by the plant's Technical Specifications (TS).

During the Palo Verde eighth Unit 3 refueling outage (U3R8) Steam Generator inspection in April, 2000, the tube at Row 31, Line 110 in Steam Generator number [EIIS:AB:SG] (SG) 3-2 was plugged due to a wear-induced indication that penetrated approximately 66% through wall, exceeding the 40% plugging limit of Technical Specification (TS) 5.5.9.4.a.6. Subsequent review of historical data revealed that this particular wear indication was present, but not identified, during the previous seventh Unit 3 refueling outage (U3R7) in October, 1998. Re-evaluation of the October 1998 SG inspection data estimates that the wear indication was approximately 45% through-wall.

Technical Specification 5.5.9 provides program controls for inspection of Steam Generator tubes to ensure that structural integrity of the RCS pressure boundary is maintained. TS 5.5.9.2 (a) identifies the requirements for tube sample selection and inspection and specifies the required actions and acceptance criteria for selected tubes. TS Table 5.5.9-2, REQUIRED ACTION C-2, states, "plug or repair defective tubes and inspect additional 2S tubes in this SG." TS 5.5.9.4.a.6 ACCEPTANCE CRITERIA identifies the plugging or repair limit as the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

On April 18, 2000, Palo Verde engineering personnel (utility non-licensed) identified that SG 3-2 was returned to service during one complete operating cycle (October 1998 to April 2000), with the tube at Row 31, Line 110 having a defect of approximately 45% through wall wear at the beginning of Cycle 8, which exceeded the 40% TS plugging limit, resulting in a condition prohibited by Technical Specifications.

II. DESCRIPTION OF STRUCTURE(S), SYSTEM(S) OR COMPONENT(S):

The purpose of the Steam Generators is to provide the interface between the reactor coolant (primary) system and the main steam (secondary) system. The Steam Generators are vertical U-tube heat exchangers, that transfer heat from the reactor coolant to the main steam system. Reactor coolant is prevented from mixing with the secondary steam by the Steam Generator tubes and the Steam Generator tube sheet, making the RCS a closed system forming a barrier to the release of radioactive materials from the reactor coolant.

The specific location of this defect (Row 31, Line 110) is at the intersection of the tube and the "batwing" support (BW1) of Steam Generator 3-2. The batwing supports restrain out-of-plane movement of the square bend region of the tube bundle in the System 80 Steam Generator design. The reduction in tube wall thickness appears to be wear-induced from normal operating conditions and is not indicative of a manufacture flaw or a corrosion-related defect. Wear-induced degradation in this region is common and defect initiation, growth and morphology are well understood. This type of wear-indication is specific to the batwing region only. The subsequent inspection results did not show this type of wear indication in tube areas outside of the batwing section.

III. INITIAL PLANT CONDITIONS:

On April 18, 2000, at approximately 1600 MST, Unit 3 was in its eight refueling outage (U3R8) with the reactor defueled, and Steam Generator inspections in progress, when APS personnel (utility non-licensed) identified that the tube at Row 31, Line 110 in Steam Generator number (SG) 3-2 had a wear-induced indication that penetrated approximately 66% through wall, exceeding the 40% plugging limit of Technical Specification (TS) 5.5.9.4.a.6. There were no other structures, systems, or components that were inoperable at that time that contributed to this event.

IV. EVENT DESCRIPTION:

During the Palo Verde eighth Unit 3 Refueling outage (U3R8) Steam Generator inspection in April, 2000, the tube at Row 31, Line 110 in Steam Generator number (SG) 3-2 was plugged due to a wear-induced indication that penetrated approximately 66% through-wall, exceeding the 40% plugging limit of Technical Specification (TS) 5.5.9.4.a.6. Subsequent review of historical data revealed that this particular wear indication was present, but not identified during the previous seventh Unit 3 refueling outage (U3R7) in October, 1998. Re-evaluation of the October 1998 SG inspection data concluded that the wear indication was approximately 45% through wall at the time of the October outage.

During the October 1998 inspection, the initial data review identified this area as having wear indication in the batwing area. However, the resolution analyst determined that there was too much noise and some distortion in the signal and recommended that this area be re-tested. Per the PVNGS inspection program a second eddy current test was performed on the entire tube. This second data set was subsequently reviewed by the analysis team, which identified a 16% wear indication at the vertical strap area but did not identify any indications at the batwing area. In accordance with establish industry guidelines, the PVNGS Inspection Program does not require any further analysis for indications less than 20%. Therefore the analysis process for this tube was considered complete.

V. SAFETY CONSEQUENCES:

A condition monitoring evaluation was performed by APS Engineering to determine if the end of Cycle 8 tube condition maintained structural integrity in accordance with the RG 1.121 required margins of safety. . The methodology of evaluation was consistent with the techniques specified in the EPRI Steam Generator Integrity Assessment Guidelines and Draft Regulatory Guide DG-1074. The components of eddy current sizing error, burst pressure relational uncertainty and material property variability were considered in the integrity evaluation. The lower bound threshold is specified as 71% through-wall for batwing wear. The tube was inspected using bobbin coil, motorized rotating pancake coil and Plus Point coil techniques. The measured depth using these techniques was a conservative estimate of approximately 66 percent through-wall, therefore structural integrity is conservatively assured and there is no safety significance with respect to this flaw for Unit 3 Cycle 8. The tube was removed from service for Cycle 9 by staking and plugging.

Based on a Unit 3 Cycle 8 condition monitoring evaluation and a Unit 3 Cycle 9 operational assessment, the potential effects associated with this as-found condition for this type of wear indication are limited to the stay cylinder batwing section of the Steam Generator. The actions taken during the April 2000 inspection also provide added assurance that a similar condition does not exist for Unit 3 Cycle 9. Additionally, inspections confirmed there is no evidence of excessive wear growth rates in the Unit 3 Steam Generators. Although this wear-initiated degradation could result in less than optimal operating conditions, there is no impact on nuclear safety.

In conclusion this event is bounded by the design basis Steam Generator Tube Rupture event in Chapter 15 of the UFSAR. There was no significant increase in the likelihood of a Steam Generator Tube Rupture Event due to the availability of mitigating systems, and the low likelihood of transient initiation as indicated by the tube integrity evaluation. The mitigation equipment required for a Main Steam Line Break (MSLB), Feedwater Line Break (FLB), or Steam Generator Tube Rupture Event (SGTR) were all available other than allowed outage times for routine maintenance and testing.

This event did not constitute a condition that would have prevented the fulfillment of a safety function as described by 10CFR50.73(a)(2)(IV) and is therefore not a safety system functional failure (SSFF).

VI. CAUSE OF THE EVENT

An investigation of this event was commenced and is being completed in accordance with the site corrective action program. Preliminary investigation results indicate the cause of this event to be a misinterpretation of data by the analysts resulting in a isolated failure to identify the tube defect. A review of previous corrective action documents, and condition monitoring evaluations did not identify any previous events where analysts failed to identify defects. Based on this review and the re-evaluation of the previous outage SG inspection data in all three Units, this event is considered to be an isolated case. There were no unusual characteristics that contributed to the missed indication such as excess noise, heat, lighting or extended work hours.

The individuals that performed the data review were appropriately trained and qualified to both industry and in-house training standards and had previous SG inspection experience at Palo Verde.

Since the investigation has not been completed at this time, a supplemental LER will be submitted if substantial information is subsequently identified that would significantly change a reader's perception of the event, or if there are substantial changes in the corrective actions described in this LER.

VII. CORRECTIVE ACTIONS:

An investigation of this event was commenced and is being completed in accordance with the site corrective action program.

The Steam Generator tube Row 31, Line 110 was staked and plugged during the eighth Unit 3 refueling (U3R8). Additional corrective actions were taken to re-evaluate the previous outage SG inspection data for the tight radius SG batwing locations in all three Units. This review was completed and no additional conditions were found that show this type of wear indication for the tight radius batwing areas.

The data management software will be revised to flag re-tests (RBD) on data (where the initial call/evaluation identified an indication) to verify the same area is re-analyzed.

This particular flaw geometry will be included for review in the site-specific data analyst training program.

VIII. PREVIOUS SIMILAR EVENTS:

No previous similar events (involving undetected wear induced degraded Steam Generator tubes) have been reported pursuant to 10CFR50.73 in the past three years.