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RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 94-004-00:on 940530,determination that secondary pressure boundary leakage existed & that applicable TS due to defective weld.Control room personnel returned unit mode 5,exited TS LCO 3.0.3 & remained in compliance.W/940627 ltr.

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Arizona Public Service Company
PALO VERDE NUCLEAR GENERATING STATION
P.O. BOX 52034 • PHOENIX, ARIZONA 85072-2034

JAMES M. LEVINE
VICE PRESIDENT
NUCLEAR PRODUCTION

192-00898-JML/BAG/KR
June 27, 1994

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Unit 3
Docket No. STN 50-530
License No. NPF-74
Licensee Event Report 94-004-00
File: 94-020-404

Attached please find Licensee Event Report (LER) 94-004-00 prepared and submitted pursuant to 10CFR50.73. This LER reports two Technical Specification Limiting Condition for Operation (TS LCO) 3.0.3 entries to restore the structural integrity of ASME Code Class 2 components to within their limit as specified in TS LCO 3.4.9. In accordance with 10CFR50.73(d), a copy of this LER is being forwarded to the Regional Administrator, NRC Region IV.

If you have any questions, please contact Burton A. Grabo, Supervisor, Nuclear Regulatory Affairs, at (602) 393-6492.

Sincerely,

James M Levine

JML/BAG/KR/rv

Attachment

cc: W. L. Stewart (all with attachment)
L. J. Callan
K. E. Perkins
K. E. Johnston
INPO Records Center

IE22
/11

LICENSEE EVENT REPORT (LER)

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TITLE (4)
TS LCO 3.0.3 Entry to Restore ASME Code Class 2 Structural Integrity

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)
0 5	3 0	9 4	9 4	0 0 4	0 0	0 6	2 7	9 4	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

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

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 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 (AREA CODE)
Burton A. Grabo, Supervisor, Nuclear Regulatory Affairs	6 0 2 3 9 3 - 6 4 9 2

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)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

On two separate occasions, at 1500 MST on May 30, 1994 and at 1542 MST on June 8, 1994, Palo Verde Unit 3 was in Mode 3 (HOT STANDBY) when Control Room personnel entered Technical Specification Limiting Condition for Operation (TS LCO) 3.0.3 following the determination that secondary pressure boundary leakage existed and that the applicable TS LCO 3.4.9 Structural Integrity ACTION statement could not be met with the Reactor Coolant System (RCS) temperature greater than 210 degree Fahrenheit (F). On May 30, 1994 and on June 8, 1994, APS ASME Section XI engineering personnel determined that the leakage at two steam generator 2 (SG-2) nozzle penetrations was unacceptable (i.e., the structural integrity was in question). For the first occurrence involving a SG-2 instrument nozzle penetration leakage, at approximately 0454 MST on May 31, 1994, Control Room personnel returned the unit to Mode 5 (COLD SHUTDOWN), exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b (i.e., RCS temperature is less than or equal to 210 degrees F). For the second occurrence involving a SG-2 sample nozzle penetration leakage, at approximately 0256 MST on June 9, 1994, Control Room personnel returned the unit to Mode 5, exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b. For both events, the apparent cause of the nozzle penetration leakage was concluded to be due to a defective weld, most likely from original fabrication. Both nozzle penetrations were repaired.

There have been no previous similar events reported pursuant to 10CFR50.73.

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I. DESCRIPTION OF WHAT OCCURRED:

A. Initial Conditions:

At 1500 MST on May 30, 1994 and at 1542 MST on June 8, 1994, Palo Verde Unit 3 was in Mode 3 (HOT STANDBY) at normal operating temperature and pressure (NOT/NOP) coming out of its fourth refueling outage.

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

Event Classification: Condition prohibited by the plant's Technical Specifications (TS).

On two separate occasions, at approximately 1500 MST on May 30, 1994 and at approximately 1542 MST on June 8, 1994, Unit 3 Control Room (NA) personnel (utility, licensed) entered TS Limiting Condition for Operation (LCO) 3.0.3 following the determination that secondary pressure boundary leakage existed and that the applicable TS LCO 3.4.9 Reactor Coolant System (RCS) (AB) Structural Integrity ACTION statement could not be met with the RCS temperature greater than 210 degree Fahrenheit (F). On May 30, 1994 and on June 8, 1994, APS ASME Section XI engineering personnel (utility, nonlicensed) determined that the leakage at two steam generator 2 (SG-2) (AB) nozzle penetrations was unacceptable (i.e., the structural integrity was in question). For the first occurrence involving a SG-2 instrument nozzle penetration leakage, at approximately 0454 MST on May 31, 1994, Control Room personnel returned the unit to Mode 5 (COLD SHUTDOWN), exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b (i.e., RCS temperature is less than or equal to 210 degrees F). For the second occurrence involving a SG-2 sample nozzle penetration leakage, at approximately 0256 MST on June 9, 1994, Control Room personnel returned the unit to Mode 5, exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b.

MAY 30, 1994 EVENT:
STEAM GENERATOR INSTRUMENT NOZZLE PENETRATION LEAKAGE

At approximately 0107 MST on May 30, 1994, Unit 3 entered Mode 3 (Cold Leg Temperature greater than or equal to 350 degrees F). At approximately 0939 MST, Control Room personnel and Operations management (utility, licensed and nonlicensed) were notifed of a steam leakage in the vicinity of the SG-2 narrow range level transmitter's (LIT) instrument nozzle penetration (SGA-LT1124A)

TEXT

which is located upstream of the level transmitter's high side root valve (RTV) (SGE-V-614). Following additional evaluation, the leakage was documented by APS ASME Section XI engineering personnel as unacceptable (i.e., the structural integrity of the SG-2 instrument nozzle penetration was in question). At approximately 1500 MST, Control Room personnel entered TS LCO 3.4.9 ACTION b (i.e., with the structural integrity of any ASME Code Class 2 component not maintained in accordance with TS SR 4.4.9, restore the structural integrity of the affected component to within its limit or isolate the affected component prior to increasing the RCS temperature above 210 degrees F). Since the RCS was above 210 degrees F, Control Room personnel entered TS LCO 3.0.3 and prepared for a plant cooldown to Mode 5 (Cold Leg Temperature less than or equal to 210 degrees F). At approximately 0454 MST on May 31, 1994, Control Room personnel returned the unit to Mode 5, exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b. At approximately 2125 MST on May 31, SG-2 was declared inoperable when it was removed from service to support nozzle repair. There were no associated TS LCO entries made nor required.

The nozzle was removed and replaced. The structural pressure boundary weld was moved to the outside of the SG shell. The structural weld was attached to an inconel weld pad that was attached to the SG shell. Both the pad and the structural weld were designed and installed per the applicable requirements of the ASME Boiler and Pressure Vessel Code. Sufficient nondestructive examinations were performed before, during, and after the repair process to provide assurance that the secondary pressure boundary was restored to within its original structural limits.

At approximately 1942 MST on June 6, 1994, a conditional release was granted for SG-2 in order to perform inservice leak test (ISLT) retest applicable for the weld repair at NOT/NOP. Control Room personnel exited TS LCO 3.4.9 ACTION b per the conditional release for the purpose of achieving the plant conditions necessary to support the prescribed retest. At approximately 2120 MST on June 7, 1994, Unit 3 entered Mode 3. By approximately 0500 MST on June 8, 1994, RCS reached NOT/NOP.

JUNE 8, 1994 EVENT:
STEAM GENERATOR SAMPLE NOZZLE PENETRATION LEAKAGE

At approximately 0854 MST on June 8, 1994, Control Room personnel and Operations management were notifed of a steam leakage in the vicinity of the SG-2 downcomer sample nozzle penetration which is located upstream of the downcomer sample isolation valve (ISV) (SGE-V-428). Following additional evaluation, the leakage was

documented by APS ASME Section XI engineering personnel as unacceptable (i.e., the structural integrity of the SG-2 sample nozzle penetration was in question). At approximately 1542 MST, Control Room personnel entered TS LCO 3.4.9 ACTION b. Since the RCS was heated above 210 degrees F, Control Room personnel entered TS LCO 3.0.3 and prepared for a plant cooldown to Mode 5.

At approximately 0256 MST on June 9, 1994, Control Room personnel returned the unit to Mode 5, exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b. At approximately 2002 MST on June 9, 1994, SG-2 was declared inoperable when it was removed from service to support nozzle repair. There were no associated TS LCO entries made nor required.

The nozzle was removed and replaced. The structural pressure boundary weld was moved to the outside of the SG shell. The structural weld was attached to an inconel weld pad that was attached to the SG shell. Both the pad and the structural weld were designed and installed per the applicable requirements of the ASME Boiler and Pressure Vessel Code. Sufficient nondestructive examinations were performed before, during, and after the repair process to provide assurance that the secondary pressure boundary was restored to within its original structural limits.

At approximately 0500 MST on June 15, 1994, a conditional release was granted for SG-2 in order to perform ISLT retest applicable for the weld repair at NOT/NOP. Control Room personnel exited TS LCO 3.4.9 ACTION b per the conditional release for the purpose of achieving the plant conditions necessary to support the prescribed retest. At approximately 1558 MST on June 15, 1994, Unit 3 entered Mode 3. By approximately 2354 MST on June 15, 1994, RCS reached NOT/NOP.

By approximately 0423 MST on June 17, 1994, both work authorization documents were completed and a final dispostion was approved and the unit was released to enter Modes 2 (START UP) and 1 (POWER OPERATION). At approximately 1519 MST on June 19, 1994, Unit 3 entered Mode 1.

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

Although there were no structures, systems, or components that were inoperable at the start of the event which contributed to this event, for both events, the apparent cause of the nozzle leakage was concluded to be due to a defective weld, most likely from original fabrication. The welds attached the nozzles (i.e.,

appurtenances) to the steam generator vessel to form the secondary pressure boundaries.

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

Not applicable - no component or system failures were involved.

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

Not applicable - no component failures were involved.

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

Not applicable - no failures of components with multiple functions were involved.

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

Not applicable - no failures that rendered a train of a safety system inoperable were involved.

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

Not applicable - there have been no component or system failures or procedural errors identified.

I. Cause of Event:

An independent investigation of this event was conducted in accordance with the APS Incident Investigation Program. In both nozzle penetration leakage events, the leakage source was not visually evident on any part of the nozzle on the outside of SG-2, and since the nozzle was attached via a partial penetration weld on the inside of the SG shell, it was suspected that the inside attachment weld was contributing to the leakage. Personnel safety concerns have precluded inspection of the inside attachment weld. Various tests (e.g., ultrasonic, liquid penetrant, or magnaflux particle tests) were conducted, however, there were no indications identifying a leak path.

On June 13, 1994, a visual examination of the inside nozzle-to-shell attachment welds on both failed nozzle penetrations was made using robotic camera/video equipment. No obvious faults were

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identified, although each weld was observed to have a small irregularity, the nature of which could not be determined. The instrument nozzle penetration exhibited a small area of build up and a leakage stain. The sample nozzle penetration exhibited a small area of steam cutting or erosion. The examination effort was not conclusive due to limited accessibility and due to the surface condition of the inside SG shell.

A preliminary evaluation, based on testing, visual examination, and restoration, has determined that the apparent cause of the nozzle leakage was due to a defective weld, most likely from original fabrication, of the inside nozzle-to-shell attachment weld which opened over time during plant operation to eventually cause the leak path (SALP Cause Code B: Design, Manufacturing, Installation Error). The visual examination using robotic camera/video equipment revealed no evidence that either of the leaks is attributed to the SG chemical cleaning process that occurred during the refueling outage.

As part of the investigation, the feasibility of performing a root cause of failure analysis of the inside nozzle-to-shell attachment welds will be evaluated. Based on the results of the feasibility study, the root cause of failure analysis may be implemented at a future outage. If the evaluation results differ from the preliminary determination, a supplement to this report will be submitted to describe the final root cause of failure.

No unusual characteristics of the work location (e.g., noise, heat, poor lighting) directly contributed to this event. There were no personnel or procedural errors which contributed to this event.

J. Safety System Response:

Not applicable - there were no safety system responses and none were necessary.

K. Failed Component Information:

Not applicable - no component failures were involved.

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

As part of the investigation, it was determined that no safety limits were violated and that the event (i.e., secondary pressure boundary leakage) is bounded by previous analyses contained in the Updated Final Safety Analysis Report Chapters 6 and 15 (RCS overcooling via secondary

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pressure boundary leaks or a main steam line break). The event did not result in any challenges to the fission product barriers or result in any releases of radioactive materials. There were no adverse safety consequences or implications as a result of this event. This event did not adversely affect the safe operation of the plant or the health and safety of the public.

III. CORRECTIVE ACTION:

A. Immediate:

Control Room personnel returned the unit to Mode 5, exited TS LCO 3.0.3, and remained in compliance with TS LCO 3.4.9 ACTION b.

B. Action to Prevent Recurrence:

As described in Section I.B, due to the recognized uncertainties of the exact state of the nozzle failure, a very conservative repair design was adopted. Since all ASME Code requirements were met for stresses and fatigue allowables over forty year design life, the repair is considered to be permanent.

IV. PREVIOUS SIMILAR EVENTS:

Although no other previous events have been reported pursuant to 10CFR50.73, a similar event occurred in Unit 2. On August 13, 1993, Unit 2 was in Mode 3 at NOT/NOP coming out of its fourth refueling outage when a steam leakage was discovered at the SG-1 pressure transmitter's (PT) instrument nozzle penetration (SGA-PT1013D) which is located upstream of the pressure transmitter's root valve (SGE-V-632). At approximately 0101 MST on August 14, 1993, Control Room personnel commenced a plant cooldown in order to repair the steam leakage. Control Room personnel did not enter TS LCO 3.4.9 ACTION b. At approximately 2334 MST on August 14, 1993, Control Room personnel returned the unit to Mode 5. SG-1 was removed from service to support nozzle repair.

The nozzle was removed and during inspection, a small steam cut area was found on the end of the nozzle. The damaged area was removed, and the nozzle was reinstalled and welded in place. At approximately 2149 MST on August 19, 1993, SG-1 was returned to service in order to perform inservice leak test (ISLT) retest applicable for the weld repair at NOT/NOP. At approximately 0528 MST on August 22, 1993, Unit 2 entered Mode 3. By approximately 0118 MST on August 23, 1993, RCS reached NOT/NOP. At approximately 2030 MST on August 29, 1993, Unit 2 entered Mode 1.

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An evaluation determined that the apparent cause of the nozzle leakage was due to weld porosity, most likely from original fabrication, of the inside nozzle-to-shell attachment weld which opened over time during plant operation to eventually cause the leak path.

At the time of the Unit 2 event, Operations and Licensing personnel determined that the TS LCO 3.4.9 pertaining to structural integrity of ASME Code Class 1, 2, and 3 components did not apply because it was located in TS Section 3/4.4 REACTOR COOLANT SYSTEM. The nozzle leakage was located within the secondary pressure boundary. On May 30, 1994, during the evaluation of the Unit 3 event, it was determined by Operations and Licensing personnel that TS LCO 3.4.9 Structural Integrity is applicable to ALL ASME Code Class 1, 2, and 3 components. Specifically TS LCO 3.4.9 ACTION b states "With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210 degrees F." This type of nozzle leakage would reveal itself at NOT/NOP. Although the TS LCO 3.4.9 ACTION statement requires restoration or isolation of the component, no action statement is provided to initiate a plant cooldown to Mode 5 in order to restore the structural integrity of the affected component. Therefore, an entry into TS LCO 3.0.3 appears to be the only action to take to meet the TS LCO 3.4.9 action statement.