Portland General Electric Company
Trojan Nuclear Plant
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Rainier, Oregon 97048
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December 9, 1992
RDM-559-92

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555

Gentlemen:

Licensee Event Report No. 92-33 is attached. This report discusses a failure to properly stress relieve a Steam Generator tube sleeve. That failure resulted in a primary to secondary coolant leakage rate exceeding the limits of Technical Specification 3.4.6.2, Reactor Coolant System Operational Leakage. A rapid Plant shutdown was commenced on November 9, 1992, with systems responding as expected.

Sincerely,

R. D. Machon
General Manager
Trojan Nuclear Plant

c: Mr. John B. Martin
Regional Administrator, Region V
U.S. Nuclear Regulatory Commission

Mr. David Stewart-Smith
State of Oregon
Department of Energy

Mr. R. C. Barr
USNRC Resident Inspector
Trojan Nuclear Plant

LER Distribution

140071

IE22-1

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Trojan Nuclear Plant	0 5 0 0 0 3 4 4	1 OF 10

TITLE (4): Failure to Properly Stress Relieve a Steam Generator Tube Sleeve Causes Primary to Secondary Leakage to Exceed Limits

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)
11	09	92	92	- 0 3 3	- 0 0	12	09	92	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	POWER LEVEL (10)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (11)			
1	0 9 5	__ 20.402(b)	__ 20.405c	__ 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	TELEPHONE NUMBER
D. B. Osborne, Compliance Engineer	503	556-5577

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	
A	A B	S G	W 1 2 0	Y							

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

ABSTRACT (16)

On November 9, 1992, while the Trojan Nuclear Plant was in Operational Mode 1 (Power Operation) at 95 percent power, a tube leak in the "B" Steam Generator was detected. The Plant was shut down due to the leak exceeding the primary-to-secondary leakage limits of Trojan Technical Specification (TTS) 3.4.6.2c, "Operational Leakage." A defect in tube Row 25-Column 17 of "B" Steam Generator was the cause of the leak. The defect was the result of the failure to stress relieve a sleeve installed in 1991 due to a personal error on the part of a contract technician to follow the installation procedure and an inadequate procedure for detecting sleeves not properly heat treated. It was determined that this event was an isolated occurrence and that other sleeves installed in the steam generators were properly heat treated. The stress relieving process and procedures implementing the process will be evaluated prior to installing additional sleeves. Several fire watches were also temporarily suspended during this event to limit the exposure of individuals to any potential radiological releases. Because this action was of limited duration, it did not present any significant safety consequences or implications. The radioactive release for this event was limited to less than 0.1 percent permitted by the TTS; and therefore, there are no safety consequences or implications associated with the event.

INTRODUCTION

Trojan Technical Specification (TTS) 3.4.6.2c, " Operational Leakage," requires that Reactor Coolant System [AB] leakage be limited to 130 gallons per day (gpd) through any one steam generator [AB, SG]. If leakage rates are exceeded, TTS 3.4.6.2 Action b requires that the leakage rate be reduced to within limits within four hours or be in Cold Shutdown within the next 36 hours. Because the "B" Steam Generator nitrogen 16 monitor indicated a leakage rate of 200 gpd, Plant shutdown was initiated and completed within the time frames required by the TTS. Accordingly, this report is submitted to fulfill the reporting requirements of 10 CFR 50.73(a)(2)(i)(A).

TTS 3.7.9, "Penetration Fire Barriers," requires a continuous fire watch for nonfunctional fire barriers. Because of potential radiological releases as a result of this event, fire watches in the release areas were temporarily suspended to protect individuals from unnecessary radiation exposure. Because suspension of the fire watch is a condition prohibited by the TTS, it is reportable under the reporting requirements of 10 CFR 50.73(a)(2)(i)(B). This report is also being submitted to fulfill those requirements. This report also satisfies PGE's evaluation, notification, and reporting obligations for defects pursuant to 10 CFR Part 21.

EVENT DESCRIPTION

At 0641 hours on November 9, 1992, while the Trojan Nuclear Plant was in Operational Mode 1 (Power Operation) at 95 percent power, the control room received a radiation alarm on Process Effluent Radiation Monitor (PERM)-26B, "Nitrogen-16 Steam Line Radiation Detector." It was ascertained that a "B" Steam Generator (S/G) tube leak had occurred. A rapid Plant shutdown was commenced at 0645 hours and an "Unusual Event" was subsequently declared. By 0747 hours, the "B" S/G was isolated and the "Unusual Event" was terminated by 1010 hours. Mode 5 was entered at 0509 hours on November 10, 1992.

At 0705 hours, fire patrol duties in the Main Steam Support Structure and Electrical Penetration Area, and continuous fire watch duties on the Control Building roof and in Area 13 (outside the Electrical Penetration shield wall) were suspended due to potential radiological releases to those areas. The fire watches were reestablished approximately 50 minutes later when Plant conditions were stabilized.

The cause of the primary to secondary leak was determined to be a defect in parent tube Row 25-Column 17 (R25C17) of S/G "B" adjacent to the lower weld expansion of the tube sleeve at the first tube support plate.

CAUSE OF EVENT

BACKGROUND

There are a total of 1018 tubes containing sleeves installed in S/Gs "B", "C", and "D". A kinetic welding process was used for the installation and is described in B&W Nuclear Services Company(BWNS) Topical Report BAW-2094PA-Revision 1, "Recirculating Steam Generators Kinetic Sleeve Qualification for 7/8 inch OD Tubes", and BWNS Field Procedure 1191345A. On December 17, 1991, a sleeve was kinetically welded to the S/G tube at the first tube support plate in R25C17 on the "B" S/G. The final step of the installation process involves stress relief heating to reduce the residual stresses induced into the sleeve and the tube at the kinetic weld. The tubes in the weld region must be stress relieved to prevent stress corrosion cracking from occurring. A heater is inserted to the proper position against a hard stop on the heater base. This places the two heater coils adjacent to the welded areas. The heater coils are then brought up to the correct temperature for a specified period of time. Stress relieving operations were completed on tube R25C17 on December 21, 1991.

Two types of heaters have been used at Trojan. They are the cartridge heater described in the BWNS Topical Report and the newer coil type heater used during the 1991 S/G tube repair. The cartridge heater is step inserted via the actuator up the S/G tube until the hard stop on the heater sheath butts against the sleeve end. The actuator exerts sufficient upward force to ensure that continuity exists between the hard stop on the heater sheath and the sleeve. The stress relief heat cycle is initiated with computer controlled monitoring and adjustments made during the process using two of the three thermocouples installed in each heater. After the process is completed the heater and tube are air cooled. Use of this heater was discontinued in 1991 because of the difficulty in removing it from the tube due to swelling of the heater. The coil heater used during the 1991 tube repair is also step inserted up the tube. However, the heater sheath contains an orifice in the hard stop area which provides an indication of air pressure characteristics as opposed to the electrical contacts contained on the sheath of the cartridge heater. As the coil heater is inserted, the contract technician verifies the distinct pressure pattern as the heater enters the sleeve and just prior to the hard stop of the heater butting against the sleeve end. Once the hard stop has been contacted and the pressure pattern has been verified, the heater is withdrawn two steps and then re-inserted three steps to re-verify the pressure pattern and to ensure sufficient upward force to prevent the heater from moving during the stress relief process. It also allows the technician to verify the pressure pattern for a second time. A pneumatic indicator is used to verify that the heater is in the proper location. The stress relief cycle is computer controlled in a manner similar to the cartridge heater.

This heater has a temperature interlock that will also terminate heater power if the upper and lower heater elements experience a large enough temperature difference between them. Such a temperature difference would indicate that the heaters were not experiencing the same heat sink, which would be the case if the heater was not fully inserted in the sleeve.

The coil heater along with other changes in field procedures are not described in BAW-2094PA, Revision 1. These changes were based on field experience with earlier tooling designs and were intended to improve the reliability of the process. Because the coil heaters are thermally more efficient than the cartridge heaters, they are controlled to a lower temperature in order to achieve the same tube temperature. The appropriate heater temperatures for this configuration were established by testing and the temperature specified in the field procedures has been specified based on this testing. BWNS concluded that these changes did not alter the critical parameters of the sleeve design and an immediate revision to the Topical report was not necessary.

DISCUSSION

The leak in tube R25C17 of S/G "B" was identified by slowly filling the secondary side with water and observing leakage into the primary side with a video camera. The flow of water identified this tube as leaking with the secondary side water level filled only to the first tube support plate.

The cause of the tube failure was accelerated corrosion from unrelieved stresses left in the tube wall after the sleeve was kinetically welded to the tube. Laboratory tests have shown that accelerated corrosion at kinetic weld locations occur if the stress relieving process is not performed. The installation procedure requires the stresses to be thermally relieved by the use of a heater inserted into the sleeve. The heater was not properly inserted into the sleeve and remained in the tube area below the sleeve. The improper placement of the heater in tube R25C17 was verified using eddy current measurements which showed altered magnetic permeability in two regions of the parent tube several inches below the tube sleeve location. Figure 1 shows the eddy current trace of the parent tube below the sleeve for R25C17 and another tube. Figure 2 shows the eddy current trace within the sleeve area of R25C17 and another sleeve which was properly heat treated. The heat treatment was then performed, but the heating occurred in the tube region below the sleeve.

The heater used in tube R25C17 was subsequently used in attempting the heat treatment of two other tubes, but could not be inserted in those tubes. It was determined that the heater was bent. Process verification testing of the heater is required by procedure to

verify proper temperature and heat distribution. The primary purpose of this testing is to demonstrate that the heater can still be used. However, since the heater was bent, it would not fit into the testing device. As allowed by procedure, a review of computer data generated from use of this heater was performed instead of the testing to determine its operability. This review indicated no malfunctions and the heater was not used again.

ROOT CAUSE

This event was caused by a combination of a personal error and an inadequate procedure.

The insertion of the heater into the proper location relied upon a contract technician observing the characteristics of the air pressure signal as the heater was inserted into the sleeve. The technician failed to detect that the heater was not positioned properly. It appears that the technician did not follow the procedure for verifying the location of the heater device. This conclusion is based on not being able to reproduce the appropriate pressure drops of the heater under laboratory conditions when inserting a bent heater (or any heater) to a location below the sleeve. Therefore, it is reasoned that the technician did not observe the proper pressure drops when the sleeve was installed in 1991.

After the bent heater was discovered, the process procedure did not contain sufficient instruction to detect that the sleeve was not heat treated. Because the bent heater could not be inserted into the process verification testing device, the procedure requires a determination that the heater had operated properly. A review of the computer data was performed, and no malfunctions were identified. It was incorrectly concluded based on the temperature indication available that the sleeve in tube R25C17 had been properly stress relieved.

The temporary suspension of several fire watches was a deliberate and prudent action to protect personnel from potential radiation exposure. After Plant conditions were stabilized, these fire watches were reestablished.

CONTRIBUTING CAUSES

The process procedure did not provide a specific range for hard stop air pressure or provide a caution about hard stop air pressure deviations from one sleeve to the next. Additionally, the hard stop air pressure for tube R25C17 was lower than the hard stop air pressures in the other sleeves in which that heater had been used, the technician failed to recognize its significance.

There was no automatic abort function on improper pneumatic indication of a hard stop position. While there is an automatic abort feature on the heater if there exists a large enough temperature difference between the two heater coils, this failed to indicate a positioning problem. Because neither heater coil was inserted into the sleeve, both heater coils experienced the same heat sink. This resulted in a temperature difference between the two coils that was not sufficient to abort the heating program.

No independent method was used to confirm that sleeves were properly heat treated.

CORRECTIVE ACTIONS

Actions Taken

Tube R25C17 in the "B" S/G was stabilized and plugged. Adjacent inservice tubes were inspected to assure no damage occurred to these tubes; the tubes were found to be acceptable.

A program was developed to evaluate and confirm by a combination of eddy current examination, record review of past heater data, and record review of past hard stop pressure data that other tube support plate sleeves installed in the Trojan steam generators received heat stress relief at the weld locations. A bobbin coil eddy current probe was used to confirm that the stress relief process did occur within the tube support plate sleeves. To provide confirmation that the heater position within the sleeves was within required tolerances to provide adequate stress relief, three criteria were developed. They are: hard stop pressure, requirements for the Heat Affected Zone(HAZ), and heater performance. Satisfying one of these criteria in conjunction with confirming that the stress relief process occurred within the tube sleeves, not in the parent tube material below the sleeves, assures that the stress relief heater was properly positioned within the sleeve. This evaluation concluded that the remaining sleeves were properly stress relieved.

The "B" S/G was slowly filled after tube R25C17 was plugged to identify other leaks. No other leakage was noted.

It was determined that the heater used on tube R25C17 was used on a total of eight tubes. These tubes were eddy current tested and found to have proper stress relief. However as a conservative measure, the remaining inservice sleeves were bobbin coil tested for indication that the stress relief occurred inside the sleeve. No other cases of failure to stress relieve were identified.

Engineering has reviewed the 1991 and 1992 B&W tube sleeve and tube plugging Non-Conformance Reports (NCRs) and found no significant safety concerns.

There were no corrective actions taken for the suspension of the fire watches, since it was judged to be a proper and prudent action.

Actions to be Taken

The stress relieving process and procedures implementing the process will be evaluated prior to installing additional Steam Generator tube sleeves. This action is due on or before March 1, 1993.

Related S/G tasks in which positioning is critical will be evaluated to determine if improvements need to be made to ensure proper positioning. This action is scheduled to be completed by March 1, 1993.

Corrective actions will be determined for providing better control over vendor processes and procedures. This action is scheduled to be completed by February 1, 1993.

SAFETY CONSEQUENCES AND IMPLICATIONS

The tube leak in the "B" S/G was detected by the PERM-26,6, and 10 radiation monitors. The plant operators acted promptly to isolate the generator and equalize Reactor Coolant System and Steam Generator pressure in accordance with the sequence and time frames specified in approved plant procedures. The radioactive release from the Plant was limited to less than 0.1 percent of the release permitted by the TTS (i. ., approximately 5E-5 mrem at the boundary of the unrestricted area). This event did not pose a significant threat to the health and safety of the public and therefore, the safety consequences are minimal.

Inspections of the remainder of the S/G sleeved tubes and review of sleeve data have concluded that the failure mechanism is unique to tube R25C17 of the "B" S/G. The sleeve design itself is such that a portion of the sleeve extends beyond the welded portions of the sleeve. This permits greater stability of the sleeved tube if there is degradation in the weld area. Although the leakage was probably in excess of the TTS limits, this event was bounded by the Steam Generator Tube Rupture analysis. The cracked S/G tube did not present any new or unanalyzed safety concern and the installed N-16 detectors, PRM-26, has proven to be able to detect small tube leaks virtually at initiation. This allows for the rapid positive identification of the leaking S/G and provides an estimate of the

leakage rate. For these reasons, the safety implications of this event are considered minimal.

The suspension of the fire watches for a brief period of time to eliminate any potential for radiation exposure to fire patrol personnel was a prudent action. There was no fire identified prior to the fire watch suspension. Because the suspension was of very limited duration, there are no safety consequences or implications for this action.

PREVIOUS SIMILAR EVENTS

No similar event involving inadequate sleeve repair due to improper stress relief was found at Trojan.

FIGURE 1

EDDY CURRENT TRACE OF PARENT TUBES
BELOW SLEEVE

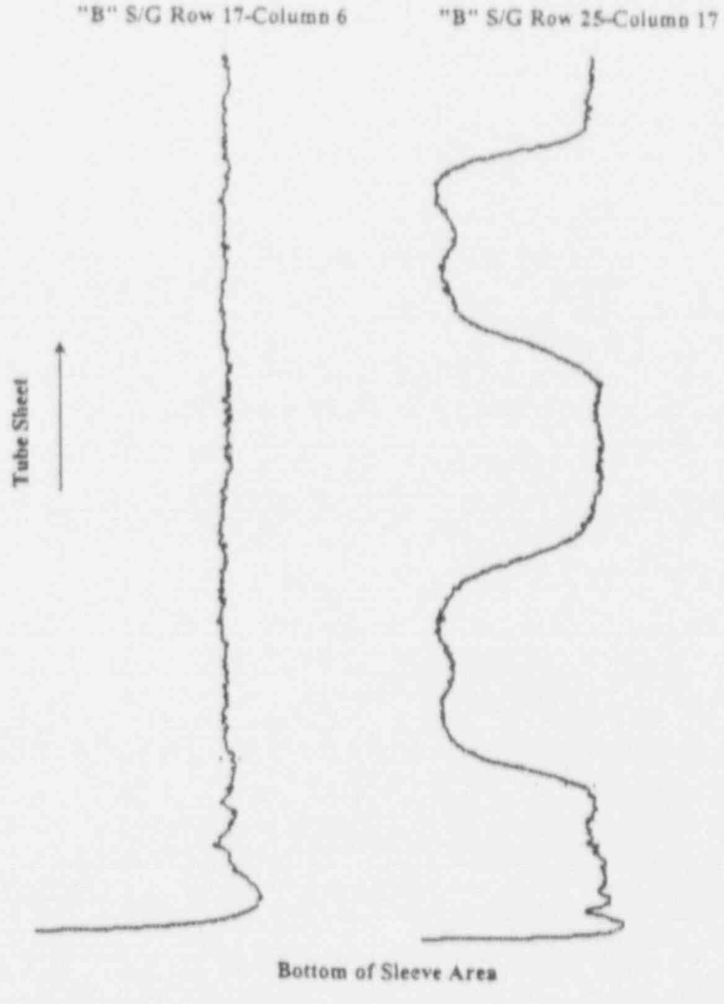
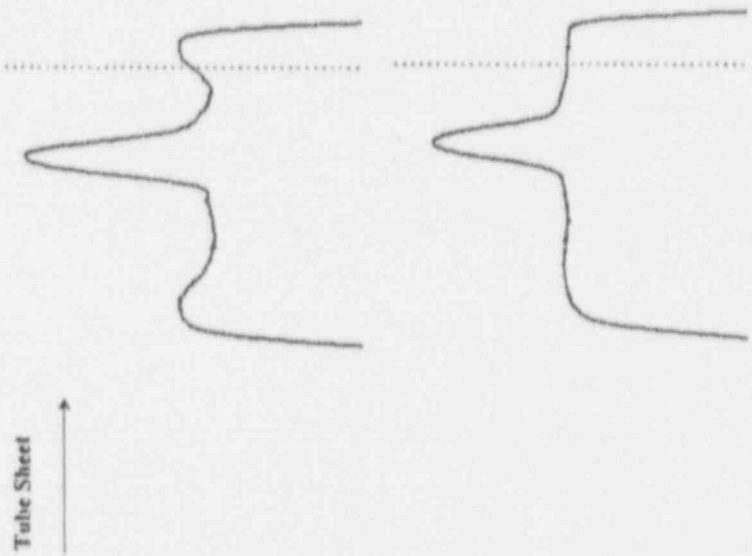


FIGURE 2

EDDY CURRENT TRACE WITHIN
SLEEVE AREA

"B" S/G Row 17-Column 6 "B" S/G Row 25-Column 17



Bottom of Sleeve Area