PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Hope Creek Operations

March 15, 1990

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

Dear Sir:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
UNIT NO. 1
LICENSEE EVENT REPORT 89-026-01

This Licensee Event Report supplement is being submitted pursuant to the requirements of 10CFR50.73(a)(2)(i) and in accordance with Revision 0 of the subject report.

Sincerely,

J.J. Hagan
General Manager -
Hope Creek Operations

RBC/

Attachment
SORC Mtg. 90-028

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LICENSEE EVENT REPORT

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
HOPE CREEK GENERATING STATION	0 5 0 0 0 3 5 4	1 OF 6

TITLE (4): REACTOR RECIRCULATION SYSTEM INSTRUMENT LINE LEAKAGE RESULTS IN MODE CHANGE TO COLD SHUTDOWN AS REQUIRED BY TECHNICAL SPECIFICATIONS - EQUIPMENT FAILURE DUE TO INSTALLATION DEFICIENCY

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	** NUMBER	** REV	MONTH	DAY	YEAR	FACILITY NAME(S)	DOCKET NUMBER(S)
1 2	3 1	8 9	8 9	- 0 2 6	- 0 1	0 3	1 5	9 0		

OPERATING MODE (9)	3	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR: (CHECK ONE OR MORE BELOW) (11)							
POWER LEVEL	0 0 0		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)
			20.405(a)(1)(iii)	XX	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
Richard Cowles, Senior Staff Engineer - Technical	6 0 9 3 3 9 3 4 3 1

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS?	\\\\\\\	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS?
B	AD	PSX	B130	N	\\\\\\\					

SUPPLEMENTAL REPORT EXPECTED? (14)	YES		NO	XX	DATE EXPECTED (15)	MONTH	DAY	YEAR	\\\\\\\\\\\\\\\\\\

ABSTRACT (16)

On 12/31/89 at 1145, with the reactor in Operational Condition 3 (Hot Shutdown), and while conducting a drywell inspection prior to unit restart, a leak from a weld on a one-inch Reactor Recirculation system elbow tap flow transmitter instrument line joint was discovered. The leakage was classified as Reactor Pressure Vessel boundary leakage, as such, Technical Specification 3.4.3.2.a was entered, which mandates that the station be in Hot Shutdown within 12 hours and Cold Shutdown (Operational Condition 4) within 24 hours. Reactor cooldown to Operational Condition 4 began at 1220 to comply with this Technical Specification.

Subsequent to the discovery of the leak, the subject instrument line was cut out and replaced with a new section of piping. The removed section was forwarded to Westinghouse Labs, Pittsburgh, for failure analysis of the pipe-to-elbow socket weld. The analysis indicates that a weld defect (crevice type geometry at the root of the weld) led to the initiation of a 0.2 inch crack at the elbow base metal interface. The primary cause has been classified as an equipment failure due to installation deficiency during original plant contruction.

At the time of discovery, other similar piping and connections in the Reactor Recirculation system were visually inspected and a sampling of similar welds were non-destructively examined to ensure pressure boundary integrity. No further problems were noted. In addition to the Westinghouse analysis, PSE&G retained Sargent and Lundy Engineering to perform an independent review the Westinghouse analysis and review existing small bore reactor recirculation system piping stresses.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Reactor Recirculation System (EIIS Designation: AD)

IDENTIFICATION OF OCCURRENCE

Reactor Recirculation System Instrument Line Leakage Results In Mode Change To "Cold Shutdown" As Required By Technical Specifications - Equipment Failure Due To Installation Deficiency

Event Date: 12/31/89
Event Time: 1145
This LER was initiated by Incident Report No. 89-185

CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 3 (Hot Shutdown), plant restart activities in progress.

DESCRIPTION OF OCCURRENCE

Prior to unit shutdown as noted in LER 89-025-00, the drywell unidentified leakage had increased from 0.15 GPM to 0.3 GPM in a two day period. As a result of this increase, a drywell inspection was initiated. On 12/31/89 at 1145, with the reactor in Operational Condition 3, and while conducting a drywell inspection prior to unit restart, a leak from a weld on a one-inch Reactor Recirculation system flow transmitter instrument line joint was discovered. The leakage was classified as reactor pressure boundary leakage, as such, Technical Specification 3.4.3.2.a was entered, which mandates that the station be in Operational Condition 3 within 12 hours, and Operational Condition 4 (cold shutdown) within 24 hours. The Senior Nuclear Shift Supervisor (SNSS, SRO Licensed) directed that control room operators begin cooldown to Operational Condition 4 at 1220 to comply with this Technical Specification.

APPARENT CAUSE OF OCCURRENCE

Analysis by Westinghouse Labs, Pittsburgh, indicates that a "crevice type geometry was seen at the root of the weld where cracking was initiated". Further, the report states, "Based on results of the evaluations, it is concluded that the observed transgranular cracking of the Hope Creek socket weld occurred by fatigue mechanism due to the cyclic loads axial to the pipe.

APPARENT CAUSE OF OCCURRENCE, CONT'D

Stress concentration at the root of the weld contributed to crack initiation.". PSE&G postulates that the weld defect (crevice type geometry) at the root of the weld generated stress concentrations which contributed to the increased piping stress levels exceeding endurance limits. The primary cause of this incident has been classified as an equipment failure due to installation deficiency during original plant construction.

ANALYSIS OF OCCURRENCE

Subsequent to the discovery of the leak, the subject instrument line was cut out and replaced with a new section of pipe and an elbow. The removed section was forwarded to Westinghouse Labs, Pittsburgh, for failure analysis of the pipe-to-elbow socket weld. Evaluations conducted included surface, non-destructive, metallographic, fractographic, and chemistry examinations.

Results of the Westinghouse analysis were forwarded to PSE&G on 3/6/90. The analysis indicates that the crevice type geometry at the root of the weld lead to the initiation of the 0.2 inch crack at the elbow base metal to weld interface. In addition to the Westinghouse analysis, PSE&G retained Sargent and Lundy (S&L) Engineering to perform an independent review of PSE&G's engineering evaluations of past small bore recirculation piping cracking. Also included in this scope was a review of data collection techniques and a request to provide recommendations for design and data collection enhancements.

Preliminary results of the S&L review were transmitted to PSE&G on 3/8/90, indicating concurrence with the Westinghouse assessment of the crack initiating mechanisms.

Adjacent welds on the line in question and the adjacent elbow tap instrumentation line were examined with no defects found. A review of installation records from original construction revealed that the subject weld was reworked following initial installation. The proximity constraints with the adjacent reactor recirculation loop header at the socket weld location may have created poor welding conditions leading to the weld defect.

In conclusion, PSE&G postulates that the weld defect (crevice type geometry) at the root of the weld generated stress concentrations which contributed to the increased piping stress levels exceeding endurance limits.

PREVIOUS OCCURRENCES

There were three previous occurrences of small bore pipe cracking in the reactor recirculation system.

The first occurrence (2/11/87, Ref: LER 87-014) was a recirc loop isolation valve seat drain interface weld failure. The section of pipe in question was removed, and replaced with a shorter piece to raise the natural resonant frequency of the piece.

The second occurrence was discovered during the station's midcycle outage in September, 1987. During inspection of the reactor recirculation system, cracks were observed at the interface welds of two of the outer elbow taps of the "B" recirc loop and cracking was observed at the "A" reactor recirc pump discharge valve bonnet and packing gland vents. As a result of these failures, all double drain valve configurations for the outer elbow taps on both loops were removed. The damaged sections were removed and replaced with increased schedule piping. Weld contours on all inner and outer elbow tap interface welds for both loops were enhanced to smooth contours to eliminate stress risers. All stem and bonnet vent valve assemblies for both the recirc suction and discharge blocking valves on both loops were removed. Instrumentation was installed to record piping strains and accelerations for ongoing evaluations. The mass reduction achieved by removing the valve assemblies reduced the corresponding piping stresses. This was correlated with recorded data and the design modification verified. Based on data available, piping stresses were within allowables and the instrumentation was removed during the subsequent outage.

The third failure occurred on 11/4/88 (Ref: LER 88-030). One of the remaining recirc isolation valve seat drain assemblies failed at the pipe to recirc valve interface. As a result of this failure, all the seat drain assemblies were removed. To continue to assess the piping system, instrumentation was reinstalled at the next refueling outage (September, 1989) but has not been made operable yet. There has been no reoccurrence of failures at past locations. As a result of the recent failure attributed to stress concentration, PSE&G will continue to monitor the piping system and is reviewing options to increase the systems' design margin and data collection techniques.

The first three occurrences were attributed to vibration induced stress resulting from cantilevered drain/vent valve configurations (extended masses).

PREVIOUS OCCURRENCES, CONT'D

It should be noted that in all past occurrences, drywell leakage monitoring did detect slight increases in unidentified leakage rates, and provided adequate time for management review and action. In none of the occurrences had the cracks progressed to a complete line failure.

SAFETY SIGNIFICANCE

As previously noted, the crack was discovered during a drywell inspection prior to a unit restart. Had this crack remained undetected and the unit returned to full power operation without repairs, the leakage into the drywell would have been detected by the leak detection system and prompted corrective action prior to having exceeded the Technical Specification integrated leakage rate limit. Any leakage of coolant into the drywell in either gaseous or liquid form would have been contained by the drywell systems as per design. Any subsequent release of this leakage through the plant radwaste systems would have been well within Technical Specification effluent limits.

Had leakage from the crack approached the unidentified leakage limit, the actions required by Technical Specifications would have resulted in plant shutdown.

Had the cracking progressed to a complete line failure, the resultant loss of coolant (LOCA) event would have been bounded by the small break LOCA analysis in the Updated Final Safety Analysis Report (UFSAR).

For these reasons, it can be concluded that the health and safety of the public were not compromised by this event.

CORRECTIVE ACTIONS

1. The subject instrument line was replaced with a new section of piping and appropriately tested prior to return to service.

2. As previously noted, the removed section of piping was forwarded to Westinghouse Labs, Pittsburgh for analysis.

3. An immediate corrective action, similar piping and connections in the Reactor Recirculation system were visually inspected, and a sampling of similar welds were NDE'd to ensure pressure boundary integrity.

CORRECTIVE ACTIONS, CONT'D

4. As previously noted, Sargent and Lundy Engineering performed an independent engineering review and concurred with the conclusions contained in the Westinghouse report. In addition, S&L has reviewed the existing stress levels based on available data. As a result of this review, it has been determined that the stress levels are within the allowable limits and continued operation is acceptable.

5. In order to provide continued assurance of small bore design adequacy, PSE&G will complete a detailed analysis to determine appropriate monitoring requirements. This review will be completed in a timely manner to support monitoring equipment installation during the stations' third refueling outage, currently scheduled for January, 1991. The enhanced instrumentation system will provide additional information to assist our staff in determining any long term system modifications and enhancements.

Sincerely,

J.J. Hagan
General Manager -
Hope Creek Operations

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SORC Mtg. 90-028