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FACIL:50-388 Susquehanna Steam Electric Station, Unit 2, Pennsylva 05000388
AUTH.NAME AUTHOR AFFILIATION
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STANLEY,H.G. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 93-009-00:on 931211,plant shut down due to drywell unidentified leakage exceeding TS 3.4.3.2 limits.Weld - overlay mod performed on cracked area & pipe support added to RBCCW RA recirculation pump HX outlet.W/940110 ltr.

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January 10, 1994

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

SUSQUEHANNA STEAM ELECTRIC STATION
LICENSEE EVENT REPORT 93-009-00
PLAS - 585 FILE R41-2

Docket No. 50-388
License No. NPF-22

Attached is Licensee Event Report 93-009-00. This report is being made pursuant to 10CFR50.73(a)(2)(i)(A), in that Susquehanna Unit 2 completed a shutdown required by the plants' Technical Specifications after unidentified Drywell leakage exceeded the Technical Specification 3.4.3.2 limit of five gallons per minute and the limit of two gallons per minute increase within any four hour time period.

H.G. Stanley
VP - Nuclear Operations

JJM/mjm

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NRC FORM 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Susquehanna Steam Electric Station - Unit 2	0 5 0 0 0 3 8 8	1 OF 0 4

TITLE (4)

Unit Shutdown Due to Drywell Unidentified Leakage Exceeding 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)
1 2	1 0	9 3	9 3	0 0 9	0 0	0 1	1 0	9 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)					
POWER LEVEL (10)	0 1 5	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
Joseph J. Meter - Power Production Engineer	7 1 7 5 4 2 - 1 1 8 7 3

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	

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 fifteen single-space typewritten lines) (16)

On December 11, 1993, at 0217 hours with Unit 2 in Condition 1 at 15% power, Unit 2 completed a shutdown required by the plants' Technical Specifications after unidentified Drywell leakage exceeded the Technical Specification 3.4.3.2 limit of five gallons per minute and the limit of two gallons per minute increase within any four hour time period. The cause of the event was due to leakage from a cracked Reactor Building Closed Cooling Water (RBCCW) pipe on the outlet from the internal heat exchanger of the 2A Reactor Recirculation Pump. The pipe crack occurred at a welded section of the pipe just above where the piping enters the 2A pump cover. The most probable cause for the pipe crack was determined to be fatigue coupled with a preferential galvanic corrosion of the weld area with respect to the base metal. In addition, the analyzed stresses were highest on the 2A outlet piping at the connection of the Reactor Recirculation Pump because of the piping support configuration. As a result of the cracked pipe, a weld - overlay of the cracked area was implemented and a pipe support was added to the RBCCW 2A Recirculation Pump heat exchanger outlet in order to reduce the stresses at the cracked area. This condition, however, did not create a significant degradation in the Station's ability to protect the health and safety of the public and/or plant personnel. The excessive Drywell leakage was not the result of primary coolant loss. The RBCCW leakage did not adversely affect the Reactor Recirculation pump/motor temperatures.

DESCRIPTION OF EVENT

On December 11, 1993, at 0217 hours with Unit 2 in Condition 1 at 15% power, the Reactor was shutdown in order to comply with Technical Specification 3.4.3.2 action statements b and e. Technical Specification 3.4.3.2.b requires the Reactor coolant system leakage shall be limited to 5 gallons per minute of UNIDENTIFIED LEAKAGE. Action statement b. states that with any reactor coolant system leakage greater than the limits in 3.4.3.2.b, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Technical Specification 3.4.3.2.e requires the Reactor coolant system leakage shall be limited to 2 gallons per minute increase in UNIDENTIFIED LEAKAGE within any four hour period. Action statement e states that with any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gallons per minute within any 4 hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

The Unit 2 Drywell leakage had been slightly elevated since July of this year. Investigation of this leakage revealed that the most likely source was from the Reactor Building Closed Cooling Water (RBCCW, EIIS Code: CC) system.

At 1730 hours on 12/10/93, a step change in Drywell Floor Drain Sump Pump level was observed by Operations (Licensed, Utility). At 1900 hours, Technical Specification 3.4.3.2 action statement e was entered when it was determined that unidentified Drywell leakage had increased by more than 2 gallons per minute in a 4 hour period. At 1920 hours, Technical Specification 3.4.3.2 action statement b was entered when the unidentified drywell leakage was determined to exceed 5 gallons per minute. Since neither four hour time limit of action statement 3.4.3.2.b or 3.4.3.2.e could be met, Unit 2 Reactor power decrease was commenced at 2020 hours on 12/10/93, the main turbine was manually tripped at 0120 hours on 12/11/93 and the Unit mode switch was placed in shutdown from approximately 15% power at 0217 hours on 12/11/93.

CAUSE OF EVENT

The cause of the event was due to leakage from a cracked RBCCW pipe on the outlet from the internal heat exchanger of the 2A Reactor Recirculation Pump (EIIS Code: AD). The pipe crack occurred at a welded section of the pipe just above where the piping enters the 2A pump cover. The most probable cause for the pipe crack was determined to be fatigue coupled with a preferential galvanic corrosion of the weld area with respect to the base metal. Contributing to the fatigue of the weld area was the RBCCW piping support configuration. Although the analyzed stresses were within acceptable limits, the stresses were highest on the 2A outlet piping at the connection of the Reactor Recirculation Pump because of the piping configuration.

REPORTABILITY/ANALYSIS

This event was determined to be reportable per 10CFR50.73(a)(2)(i)(A), in that Susquehanna Unit 2 completed a shutdown required by the plants' Technical Specifications after unidentified Drywell leakage exceeded the Technical Specification 3.4.3.2 limit of five gallons per minute and the limit of two gallons per minute increase within any four hour time period. This condition, however, did not create a significant degradation in the Station's ability to protect the health and safety of the public and plant personnel. The excessive Drywell leakage was not the result of primary coolant loss and the RBCCW leakage did not adversely affect the Reactor Recirculation pump/motor temperatures.

In accordance with guidance provided in NUREG 1022, Supplement 1 item 14.1 and 10CFR50.4(d), the required submission date for this report was determined to be 01/10/94.

CORRECTIVE ACTION

Prior to the Unit shutdown (July 1993), elevated drywell leakage had been noted and Operations and Systems Engineering personnel monitored drywell leakage, RBCCW head tank fill rates and Reactor Recirculation pump/motor temperatures. As a result of this monitoring, the suspected cause of the elevated Drywell leakage was an RBCCW line or valve near the 2A Recirculation pump area. Procedural enhancements were then added to more adequately respond to this leakage. Also, temperature monitoring performed during this time frame showed no increase in applicable Recirculation pump or motor temperatures.

On 12/11/93, as a result of unidentified Drywell leakage exceeding Technical Specification limits, Unit 2 was shutdown. A containment walkdown was performed to determine the source of the drywell leakage. The major source of the leakage was determined to be the cracked RBCCW pipe. Several small leaks were also found on other RBCCW piping inside containment. These leaks were on the order of several drips per minute as compared to gallons per minute from the cracked pipe. Action was taken to reduce this leakage by ensuring valves closed fully, tightening pipe caps and initiating the appropriate work documents.

As a result of the cracked pipe, a weld - overlay modification was performed on the cracked area. In addition, a pipe support was added to the RBCCW 2A Recirculation Pump heat exchanger outlet in order to reduce the stresses at the cracked area. The remaining RBCCW piping for the Unit 2 Recirculation pumps as well as the Unit 1 piping was reviewed to determine if additional pipe supports were needed. No additional pipe supports were needed on either unit.

ADDITIONAL INFORMATION

Failed Component Identification: N/A

Past Similar Events:

A review of past Licensee Events Reports (LERs) for the station identified no previous LER's in which a Unit shutdown was completed due to elevated Drywell leakage.

The following is a list of past LER's in which a Unit shutdown was completed as required by the plant's Technical Specifications:

Docket No. 50-387, LER 84-045-00	Scram Discharge Volume Vent/Drain Valve Surveillance Completed Late.
Docket No. 50-387, LER 84-048-00	Containment Purge Valves Found With Excessive Leakage After Containment Inerting.
Docket No. 50-387, LER 86-021-00 No. 50-388	Unit 1 and 2 Shutdown Due to Inoperable Emergency Service Water System.
Docket No. 50-387, LER 87-014-00	Containment Purge Valves Found With Excessive Leakage.
Docket No. 50-387, LER 89-006-00	Manual Shutdown Due to Inoperable Vacuum Breaker.