NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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.

FACILITY NAME (1): Washington Nuclear Plant - Unit 2

DOCKET NUMBER (2): 0 5 0 0 0 3 9 7

PAGE (3): 1 OF 0 9

TITLE (4): Degradation of the Primary Containment Pressure Boundary Caused a Plant Shutdown Due to Cracks on High Pressure Core Spray Small Bore Piping

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 0	2 3	9 0	9 0	0 2 8	0 0	1 2	0 4	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	X	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)	X	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: S. L. Washington, Compliance Supervisor

TELEPHONE NUMBER: AREA CODE 5 0 9 — 3 7 7 - 2 0 8 0

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
B	B G	P S F		Y					

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)

Beginning on October 23, 1990, four related reportable events or conditions occurred. First, on October 23, 1990 a hairline crack was discovered on a small bore drain line pipe off of the High Pressure Core Spray (HPCS) Suppression Pool Test Return Line. This condition was considered a degradation of the Primary Containment. Second, on October 31, 1990 the HPCS System, a single train safety system, was declared inoperable due to a linear indication on a small bore pipe (weld) attached to the HPCS Injection Line. Third, on November 2, 1990 at 1630 hours Plant Engineers determined that the linear indication was a crack (not a through the pipe wall crack). At the time this condition was considered a more significant degradation of the Primary Containment pressure boundary and at 1726 hours a Plant shutdown was initiated and an Unusual Event declared. The Plant was manually scrammed at 2153 hours. The fourth reportable event occurred when a Reactor Protection System (RPS) actuation occurred due to a "Low" Reactor water level trip. At the time of the event Plant Operators were reducing Reactor pressure so that water could be fed to the Reactor using a Condensate Booster Pump.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

On October 23, 1990 immediate corrective action was taken to isolate the drain line from both Primary Containment and the HPCS System. Other HPCS small bore pipe welds were examined. Engineering analysis determined that the linear indication found on the drain line attached to the HPCS Injection Line on October 31, 1990 would not affect the integrity of the Primary Containment pressure boundary.

The root cause of the HPCS Injection Line pipe crack event is indeterminate in that the cause for the initiation of the crack can not been determined. The root cause of the Test Return Line vent and drain line cracks is believed to be fatigue. The root cause of the RPS actuation due to the Low Vessel Water Level event is performance based in that the Reactor mass input/output was not balanced.

Corrective Actions include replacing the HPCS Injection Line drain line pipe, and redesign and replacement of the Test Return Line drain and vent line connections. Regarding the RPS actuation, Plant Operations Management is reviewing the event with each each Operations Crew.

Plant Conditions

a) Power Level - 100%
b) Plant Mode - 1 (Power Operation)

Event Description

Beginning on October 23, 1990, four related reportable events or conditions occurred.

On October 23, 1990 at 0513 hours, a Plant Equipment Operator (non-licensed) discovered a hairline crack in a socket weld joining a small bore (3/4") pipe drain line (HPCS-V-36 drain valve) to the 12" HPCS Suppression Pool Test Return Line. The crack was discovered because it was a through the wall crack and a small amount of water was leaking through the crack. At the time of this discovery, the Equipment Operator was closing the Suppression Pool Test Return Line Manual Isolation Valve (HPCS-V-64) due to the inoperability of the Test Return Line Automatic Isolation Valve (HPCS-V-23)(See LER 90-25). Closing HPCS-V-64 isolated the cracked pipe from the Primary Containment pressure boundary.

At 0850 hours, the HPCS Suppression Pool Test Return Automatic Isolation Valve, HPCS-V-23, was closed and tagged to prevent it from being opened. This isolated the cracked line from the HPCS System.

An investigation was initiated by Plant Management to review other HPCS small bore piping welds which had previously been identified as having a some probability of failure due to HPCS System vibration and pipe configuration.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

On October 31, 1990, during nondestructive examinations, using magnetic particle and dye penetrant techniques, Supply System Engineers found a linear indication on the socket weld joining the small bore (3/4") piping for drain valves HPCS-V-21 and HPCS-V-22 to the 12" HPCS Injection Line. A Plant Operating Committee (POC) Immediate Disposition was approved to allow continued operation, because a linear indication is not necessarily an indication of a crack, there was no leakage at reactor pressure (approximately 1000 psi), and engineering analysis showed that the drain line would remain intact during and after a design basis seismic event and a Loss of Cooling Accident (LOCA). Since the affect of operating the HPCS System on the pipe could not be characterized by engineering analysis the HPCS System was declared inoperable and all HPCS pump starts disabled.

Also on October 31, 1990, a second linear indication was found on the small bore pipe connection for a 3/4" vent line attached to the HPCS Suppression Pool Test Return Line directly above the drain line found cracked on October 23, 1990. No action was required since HPCS-V-64 and HPCS-V-23 were previously closed isolating this section of pipe from the Primary Containment pressure boundary and from the HPCS System.

On November 2, 1990 at 1630 hours Supply System Engineers, using ultrasonic nondestructive testing techniques determined that the linear indication found on the drain line attached to the HPCS Injection Line was a crack. Since Primary Containment integrity could not be assured, at 1726 hours a Plant Shutdown was initiated and an Unusual Event Declared due to a Technical Specification (Containment Integrity 3.6.1.1) forced shutdown.

Plant operators using the General Operating Procedure (PPM 3.2.1), Normal Shutdown to Cold Shutdown, at 2153 hours manually scrammed the reactor. At the time of the scram, reactor power was 20%, the Main Turbine/Generator was off-line, and reactor vessel level was being controlled by the Startup Level Controller (RFW-LIC-620) using the Startup Flow Control Valves (RFW-FCV-10A and 10B). When the reactor was scrammed the vessel level dropped rapidly due to void collapse and then recovered as inventory accumulated. At 2157 hours a Reactor Vessel Water Level "High" (Level 8) trip occurred at +54 inches. This, by design, caused the Reactor Feedwater Drive Turbine (RFW-DT-1A) to trip which in turn caused a the operating Reactor Feedwater Pump (RFW-P-1A) to shut down.

At the time of the Feedwater Turbine trip Reactor pressure had decayed to approximately 650 psig which was below the pressure control setpoint of the Digital Electro-Hydraulic (DEH) (Main Turbine) Control System and the Main Turbine Bypass Valves were closed. Over the next 26 minutes, reactor pressure slowly increased from 650 psig to 700 psig and water level slowly decreased to +41 inches. At 2223 hours Plant Operators began to reduce Reactor pressure by decreasing the DEH pressure control setpoint which opened the Main Turbine Bypass Valves. The Operators were reducing reactor pressure to approximately 600 psig so that water could be added to the Reactor Pressure Vessel (RPV) using a Condensate Booster Pump. Opening

the Bypass Valves allowed the Reactor water level to decrease along with Reactor pressure. Just prior to the event Plant Operators increased the DEH depressurization rate from 16 psig/minute to 25 psig/minute. At 2231 hours, when reactor pressure reached 620 psig and reactor vessel level had dropped to +15 inches, feedflow was established to the vessel and level began to increase. However, at approximately that same time changes in the Feedwater Startup Flow Control Valve (FCV) position caused the level to decrease again and a Reactor Vessel Low Level (+13 inches) trip occurred at 2234 hours. The changes in the Startup FCV are attributed to the change in the deprressurization rate.

On November 3, 1990 at 0856 hours, the Unusual Event terminated as the Plant reached Operational Mode 4 (Cold Shutdown).

Immediate Corrective Action

The immediate corrective actions taken during the event are included in the event description above. There were no immediate corrective action associated with the RPS actuation because there was no actual Control Rod movement since they were already fully inserted into the core, and Reactor pressure had decreased to the point where makeup flow could be provided by the Condensate System.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is being reported per the requirements of four 10CFR50.73 criteria.

   Per 10CFR50.73(a)(2)(i)(A) as a completion of a Plant Shutdown required by Technical Specifications. Reported verbally per 10CFR50.72(b)(1)(i)(A) at 1742 hours on November 2, 1990.

   Per 10CFR50.73(a)(2)(ii) as a condition that seriously degraded a primary safety barrier (Primary Containment). Reported per 10CFR50.72(b)(1)(ii) at 0612 hours on October 23, 1990 and at 1742 hours on November 2, 1990. The second event reported, the crack found on the drain line on the HPCS Injection Line identified on November 2, 1990, was later downgraded to not reportable when it was determined by engineering analysis that the line would not have failed during normal or accident conditions.

   Per 10CFR50.73(a)(2)(iv) as an automatic actuation of the Reactor Protection System. Reported per 10CFR50.72(b)(2)(ii) at 2317 hours on November 2, 1990.

   Per 10CFR50.73(a)(2)(v) as an a condition which prevented a safety system from accomplishing its function. The HPCS System is a single train system which was taken out of service during this event. Reported per 10CFR50.72(b)(2)(iii)(D) at 0015 hours on November 1, 1990.

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2. There were no structures, systems, or components inoperable prior to this event which contributed to the event except for mechanical problems with the Feedwater regulator and opening pressure booster on RFW-FCV-10A/B. These problems had previously been identified and may have contributed to the Low Vessel Water Level RPS actuation event. As discussed in LER 90-25, the HPCS Automatic Isolation Valve, HPCS-V-23, was inoperable during this event, it did not contribute to this event.

3. Upon shutdown of the plant, the drain line pup piece was removed from HPCS-V-21 for metallurgical evaluation. The linear indication was cut open to expose the fracture surface. It was determined upon examination that the crack was caused by fatigue. Subsequent scanning electron microscopy (SEM) examination determined the fatigue was due to intermittent loading, based on the presence of "beach" marks. This initialized a testing program on the HPCS system to define loading conditions at the drain line location. The HPCS-V-21 and HPCS-V-22 drain line and HPCS Injection Line directly above the drain connection was instrumented with five accelerometers. Three accelerometers were mounted (tri-axially) on the large bore piping and two accelerometers (horizontal plane) were mounted on HPCS-V-22. Utilizing the described instrumentation, five system tests were performed as follows: 1) HPCS surveillance, PPM 7.4.5.1.11, simulating throttling HPCS-V-23 in the Suppression Pool Test Return Line; 2) HPCS injection to the RPV (stroking open the HPCS Injection Valve (HPCS-V-4) under full pump differential pressure; 3) HPCS-V-4 LLRT utilizing a positive displacement hydro pump simulating yearly 950 psid leakage testing; 4) HPCS-V-4 surveillance testing involving valve stroking while simulating the RPV at rated pressure; 5) Static stroking of HPCS-V-4.

   The results from this dynamic testing revealed that significant loads are generated from the RPV injection event when the HPCS-V-4 valve begins opening with full pump discharge pressure differential across it. These loads were then digitized and applied to a dynamic model of the HPCS-V-21/22 drain line to establish a stress field for subsequent fracture mechanics analysis.

   A fracture mechanics model was developed using the actual crack configuration. The stresses developed by the HPCS injection to the RPV were used as input into the model. Based upon the results of the number of cycles developed during the HPCS injection and comparing the crack growth between "beach" marks, it was determined that intermittent crack growth was due to HPCS injections. Therefore, correlation between actual system loads and measured crack propagation was established. One indeterminate factor exists dealing with the crack initiation event. Computer modeling assumes a crack has been initiated by some unknown mechanism. (Possibly an initial construction defect.)

Engineering analysis of the HPCS system tests concluded that the number of HPCS injections incurred to date would not have initiated the crack, but rather would have only propagated the crack initiated by some other mechanism. Further, analysis has shown that roughly an additional 60 HPCS injections would be required to propagate an initiated crack into a through-wall fatigue crack.

4. Concurrent with the fracture mechanics and HPCS system tests, liquid penetrant examinations were being conducted on selected vent and drain line connections which performed a Containment integrity function or were within a small break LOCA boundary which could not be isolated from the RPV, and were contained in pump driven systems. Between October 23, 1990 and November 8, 1990 104 fillet welds on 40 vent/drain line connections were nondestructively examined. After the results of the fracture mechanics analysis, the scope of NDE examination was narrowed to include all HPCS, Low Pressure Core Spray System(LPCS), Residual Heat Removal System (RHR) Low Pressure Core Injection Mode, and Reactor Core Isolation Cooling (RCIC) injection line connections immediately downstream of the injection isolation valves which open against full pump differential pressure. No indications were found using fluorescent liquid penetrant nondestructive testing techniques.

5. Plant Operations Management has evaluated the Operations Crew management of reactor pressure and water level following the manual scram. The RPS actuation was due to the establishment of an RPV mass removal mechanism (steam flow through the bypass valves) which exceeded the inservice makeup capability of the Control Rod Drive System inflow. It was assumed an RPV feed source of sufficient capacity would become available prior to reaching the RPS "Low" water level setpoint. A combination of the Feedwater Startup FCV control speed and level fluctuations frustrated determination of level margin available to preclude actuation.

6. The root cause of the pipe crack on the drain line attached to the HPCS Injection Line is indeterminate. Engineering analysis has established the cause of the crack propagation but no event analyzed would have been sufficient to initiate the crack. Therefore, it is believed that the flaw has existed since the original weld was made. The root cause evaluation of the cracks on the vent and drain lines on the Test Return Line is not yet complete; however, they are believed to be fatigue cracks. If the final root cause is different a Supplemental LER will be sent.

7. The preliminary root cause of the RPS actuation event is the situation analysis by the Operating Crew was less than adequate in that they did not balance the mass input and output to the RPV. If the final root cause is different a Supplemental LER will be sent. A contributing cause was the mechanical problems associated with RFW-FCV-10A/B.

B. Further Corrective Action

1. The HPCS-V-21 and HPCS-V-22 drain line was replaced. Further, these welds will be redesigned and replaced during the next Refueling Outage.

2. Both the HPCS-V-36 drain line and the HPCS-V-74 vent line have been redesigned and replaced.

3. A plant modification is planned to reduce the vibration in the HPCS Suppression Pool Test Return Line. The design is scheduled to be completed by June 1991 and installation is planned for Refueling Outage R7 (Spring 1992).

4. Further evaluations will be conducted to determine if other small bore pipe configurations should be modified, and if other design modifications need to be made to reduce the vibration or loading on these small bore pipes. When appropriate, nondestructive testing will be performed until the above evaluations and modifications have been completed.

5. Corrective maintenance was performed on RFW-FCV-10A/B.

6. Plant Operations Management is reviewing this event with each of the six Operation Crews. The discussion focuses on Management's expectations with regards to Reactor pressure and water level control strategies.

Safety Significance

There is a minimal safety significance risk associated with the cracks on the vent and drain lines attached to the HPCS Suppression Pool Test Return Line. It is the opinion of Plant Engineers that without planned Test Return Line modifications both of these lines would have eventually failed. The Test Return Line redesign is scheduled to be completed by June 1991 and installation is planned for Refueling Outage R7 (Spring 1992). If the vent or drain line did fail it most likely would occur during HPCS testing when the high vibration condition occurs. If the failure did occur during testing, there is no safety significance because the lines can be isolated. Also, it is unlikely that failures would occur during accident conditions because the HPCS Suppression Pool Test Return Automatic Isolation Valve HPCS-V-23 closes whenever the HPCS System is initiated and there would be no flow in the line. There is no safety significance associated with the crack found in the HPCS-V-21/22 drain line which was determined by engineering analysis to be capable of withstanding 60 more cycles before the line would have failed. This 60 cycles bounds the expected number of times this event

would be expected to occur during the remaining Plant License lifetime. There is no safety significance associated with the HPCS System being out-of-service. The length of time the System was out of service was within the Technical Specification allowable HPCS outage time. Further, the RCIC System (non-safety system) was available for high pressure water injection to the RPV. The Automatic Depressurization System (ADS) whose function is to depressurize the RPV in the event that no high pressure source of injection water is available was also available. There is no safety significance associated with the RPS actuation since all Control Rods were already full inserted prior to the event and reactor water level was promptly recovered.

Similar Events

LER 89-15 describes an event where a vent line broke off of the HPCS Suppression Pool Test Return Line. Corrective actions committed to in LER 89-15 are still being implemented. Installation of a design modification to reduce vibration on the HPCS Suppression Pool Test Return Line is planned for Refueling Outage R7 in the Spring of 1992. Both the small bore pipes found cracked had previously been identified as high vibration locations. Both welds were examined during the Refueling Outage R4 (spring 1989) with no indications identified. LER 85-11-00 and LER 85-11-01.

There have been several events associated with level fluxuations at WNP-2 (LERs 86-038, 87-002, 88-001 and 88-003). However, none of these LERs were associated with events specific to control of level during a controlled shutdown.

EIIS Information

Text Reference	EIIS Reference System	Component
High Pressure Core Spray System(HPCS)	BG	-
HPCS Suppression Pool Test Return Line	BG	PSP
Primary Containment	C	-
HPCS Injection Line	BG	PSP
Reactor Protection System	JC	-
Condensate Booster Pump	SD	P
HPCS Drain Valve 36 (HPCS-V-36)	BG	V
HPCS Supression Pool Test Return Automatic Isolation Valve (HPCS-V-23)	BG	V
HPCS Suppression Pool Test Return Manual Isolation Valve (HPCS-V-64)	BG	V
HPCS Drain Valve 21 (HPCS-V-21)	BG	V
HPCS Drain Valve 22 (HPCS-V-21)	BG	V
HPCS Pump	BG	P
Main Turbine/Generator	TA/TB	TRB/GEN

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Startup Feedwater Level Controller (RFW-LIC-620)	JB	LIC
Feedwater Startup Flow Control Valve (RFW-FCV-10A/B)	SJ	FCV
Reactor Feedwater Drive Turbine 1A (RFW-DT-1A)	SJ	TRB
Reactor Feedwater Pump 1A (RFW-P-1A)	SJ	P
Digital Electro Hydraulic System (DEH)	JJ	-
Main Turbine Bypass Valves	SO	V
Control Rod	AA	-
Condensate System	SD	-
Reactor Pressure Vessel((RPV)	AC	-
HPCS Injection Valve (HPCS-V-4)	BG	V
HPCS Vent Valve (HPCS-V-74)	BG	V
Low Pressure Core Spray System (LPCS)	BM	-
Residual Heat Removal (RHR)	BO	-
Reactor Core Isolation Cooling (RCIC)	BN	-
Automatic Depressurization System (ADS)	BG	-