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ARBUCKLE,J.D. Washington Public Power Supply System
BAKER,J.W. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 91-030-00:on 911104,small leak noted in welded connection between RHR sys drain valve & RHR Loop A shutdown cooling return valve.Caused by const fabrication defect.RHR valve V-161A isolated from RCS.W/911127 ltr.

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TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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November 27, 1991
G02-91-216

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 91-030

Dear Sir:

Transmitted herewith is Licensee Event Report No. 91-030 for the WNP-2 Plant. This report is submitted in response to the report requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Very truly yours,

LT Harold for

J. W. Baker
WNP-2 Plant Manager

JWB:ac

Enclosure:
Licensee Event Report No. 91-030

cc: Mr. John B. Martin, NRC - Region V
Mr. C. Sorensen, NRC Resident Inspector (M/D 901A)
INPO Records Center - Alanta, GA
Ms. Dottie Sherman, ANI
Mr. D. L. Williams, BPA (M/D 399)
NRC Resident Inspector - walk over copy

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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): Washington Nuclear Plant - Unit 2

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

PAGE (3): 1 OF 0 6

TITLE (4): Technical Specification Required Plant Shutdown due to Reactor Pressure Boundary Leakage Through Defective Weld on Residual Heat Removal System Drain Line 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 1	0 4	9 1	9 1	0 3 0	0 0	1 1	2 7	9 1		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 3

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)
J. D. Arbuckle, Compliance Engineer	5 0 9 3 7 7 - 4 1 4 5

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)

		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 November 4, 1991 at 0423 hours, an inspection team that was performing a routine 1000 psig drywell inspection during startup from an outage, informed the Shift Manager that a small leak (approximately 20 drops per minute) had been identified in a welded connection between a Residual Heat Removal (RHR) System drain valve and the RHR, Loop A, Shutdown Cooling return line. The Shift Manager determined that this condition represented pressure boundary leakage.

Although the leakage could have been isolated by manually closing an RHR isolation valve, the decision was made by Plant Management to declare an Unusual Event and commence a reactor shutdown because of the leakage and the extended length of time required to close the valve and isolate the leakage from the reactor pressure coolant boundary. At 0454 hours an Unusual Event was declared and Plant Control Room Operators commenced shutdown of the Plant. As immediate corrective actions, the Plant was manually scrammed at 0525 hours and, at 0608 hours, the manual isolation valve was closed to isolate the leakage. At 0618 hours, the Unusual Event was terminated.

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

Abstract (Cont'd)

The cause of this event was a construction fabrication defect in a fillet weld at a 3/4-inch sock-o-let which makes up a drain connection to drain valve RHR-V-161A. Further corrective actions consisted of 1) removing the drain connection and replacing it with a short-couple nipple and pipe cap, and 2) modifying three other connections located in the reactor pressure boundary that were identified by Engineering as being susceptible to failure.

This event posed no threat to the health and safety of either the public of Plant personnel.

Plant Conditions

a) Power Level - 3%
b) Plant Mode - 2 (Startup)

Event Description

On November 4, 1991 during a routine inspection of the Reactor Drywell in preparation for Plant re-start following an outage to repair a Main Condenser tube leak, Plant personnel discovered a small leak in Loop A of the Residual Heat Removal (RHR) System. At 0300 hours, an inspection team consisting of the Shift Support Supervisor, an Equipment Operator and a Health Physics Technician entered the drywell to perform a routine startup 1000 psig inspection. At 0423 hours, the inspection team informed the Shift Manager that a pinhole leak, with an estimated volume of 20 drops per minute, had been identified in a welded connection between drain valve RHR-V-161A and the RHR, Loop A, Shutdown Cooling return line. The weepage-type leak was emanating from a fillet weld at a 3/4-inch sock-o-let which makes up a nipple connection to drain valve RHR-V-161A.

The Shift Manager determined that this condition represented pressure boundary leakage and entered the appropriate Technical Specification Action Statement for the Reactor Coolant System, which required the Plant to be in Hot Shutdown within 12 hours and Cold Shutdown within the next 24 hours. Pressure boundary leakage is defined in the WNP-2 Technical Specifications as "leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel."

Although the leakage could have been isolated by closing manual isolation block valve RHR-V-112A, a decision was made by Plant Management to declare an Unusual Event and commence a reactor shutdown because of the leakage and the extended length of time it would take to manually close RHR-V-112A to isolate RHR-V-161A from the reactor pressure coolant boundary. Isolation valve RHR-V-112A is a 12-inch valve in the drywell that is locked open. At 0454 hours an Unusual Event was declared in accordance with procedures and Plant Control Room Operators commenced shutdown of the Plant. The Plant was manually scrammed at 0525 hours and, at 0608 hours, RHR-V-112A was closed which isolated RHR-V-161A from the reactor coolant system. The Unusual Event was then terminated at 0618 hours.

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

Immediate Corrective Action

Plant Operators took appropriate and timely action to shutdown the Plant and isolate RHR-V-161A from the reactor coolant system.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is reportable in accordance with 10CFR50.73(a)(2)(i)(A), "The completion of any nuclear plant shutdown required by the plant's Technical Specifications." The WNP-2 Technical Specifications do not permit any reactor coolant pressure boundary leakage.

2. There were no other structures, systems or components that were inoperable prior to the event that contributed to the event.

3. The cause of this event is a construction fabrication deficiency in the fillet weld at the process pipe socket forging (3/4-inch sock-o-let) which makes up the nipple connection to drain valve RHR-V-161A. The RHR-V-161A drain line connection, which is located in a low-vibration area, failed due to an internal lack of fusion on the inside of the weld. The defect, an estimated 1/4-inch area, had propagated through the weld and resulted in a through-wall crack, presumably after several years of low-level vibration. This weld was completed during Plant construction.

4. Although the failure mode of this particular connection was unique, there have been 26 fatigue-related weld failures recorded at WNP-2 prior to this event. Twenty of those failures were on drain, vent and test lines at the point where they connected to large bore piping. Only one previous failure, attributed to fatigue, was initiated by a construction weld defect in a hydraulic line to the Reactor Recirculation (RRC) System flow control valves (not a drain, vent or test line configuration). With the exception of one other failed drain connection that was due to an inadequate weld repair, all other failures were due to fatigue cracking initiating from the toe of the weld. In the majority of these cases, the failed lines were on or near piping that exhibited high vibration.

5. A systematic method of implementing design improvements for piping failures was implemented in July of 1987. Extensive efforts have been ongoing to preclude repetitive failures as well as any new failures whenever possible. The focus of these design improvements is to eliminate the cause of these failures through improved design configuration and minimizing the source of cyclic loading.

As a result of a previous event (LER 90-028), the program for reducing piping failures was modified to also include 85 test/vent/drain connections located in systems which were susceptible to flow-induced vibration. It was recommended that the connections be modified at a rate of five each year, and inspections be performed each year on the unmodified connections.

The failed RHR-V-161A drain connection described in this LER was included in the program scope for reducing piping failures, and the schedule was to strengthen the connection by replacing the fillet weld with a butt weld during the Spring, 1991 maintenance and refueling outage.

However, when the work was started it was discovered that a good butt weld could not be installed, due to the restricted area in which the connection was located, in the time allotted by the ALARA review. Therefore, several options were discussed and the decision was made to redesign the line configuration by changing out the existing valves and installing lighter short-coupled valves. Accordingly, a Fluorescent Liquid Penetrant examination was performed on the RHR-V-161A fillet weld connection and no defects were identified. Fillet welds can only be surface inspected and the geometry of the weld does not allow for meaningful readings from radiography. As a result, Fluorescent Liquid Penetrant nondestructive examination was the most sensitive method available for identifying fatigue cracking in this type of weld. Because no defects were identified, the decision was made to defer the work on this connection until the 1992 maintenance and refueling outage to allow for redesign of the line and procurement of parts.

B. Further Corrective Action

1. As an interim action, the failed RHR-V-161A drain connection was removed and replaced with a short-couple nipple and pipe cap. Furthermore, the drain connection will be redesigned and replaced during the 1992 maintenance and refueling outage.

2. An evaluation was performed to determine if any additional interim actions were needed pertaining to drain connections. As a result of the evaluation, drain connections HPCS-V-37/38 and RHR-V-157B/158B were also removed and replaced with a short nipple and pipe cap. In addition, the RHR-V-163/164 fillet weld (socket connection at process line valve RHR-V-113) was contoured (changing the weld profile) to add strength. The additional connections were chosen based on an engineering evaluation as being susceptible to vibration and which are located within the reactor pressure boundary where a small break LOCA potential exists. These drain connections will also be redesigned and replaced during the 1992 maintenance and refueling outage.

3. Further evaluations will continue to be performed, as part of the program for reducing piping failures, to determine if design modifications need to be made to reduce vibration or loading. When appropriate, nondestructive examination will also continue to be performed on those piping configurations.

Safety Significance

There are no unacceptable consequences associated with this event. Plant Operators took appropriate and timely action to shutdown the Plant and isolate RHR-V-161A from the reactor coolant pressure boundary. Furthermore, the characteristic of the 1/4-inch defect was such that it would have propagated around the outside of the weld and, therefore, represented a "leak-before-break" condition. However, the Plant Leak Detection System would have detected the leakage and subsequent primary coolant system pressure boundary degradation. The Leak Detection System sensitivity and response time is such that an unidentified leakage rate increase of one gpm in less than one hour will be detected. Therefore, corrective action would be taken well before the integrity of the nuclear system process barrier could be significantly compromised. In addition, if the problem had remained undetected and the connection failed completely during operation (thereby representing a small-break LOCA), the resulting high drywell pressure condition would have alerted Plant Operators to take appropriate actions. A small-break LOCA is also bounded by the WNP-2 safety analysis. Accordingly, this event posed no threat to the health and safety of either the public or Plant personnel.

Similar Events

1. LER 85-011-00/85-011-01, "Pressure Boundary Leakage." This LER described an event where two 3/4-inch RHR lines were discovered to be leaking due to small cracks. The lines were repaired and ten additional lines were examined. No further discrepancies were identified.

2. LER 89-015, "High Pressure Core Spray (HPCS) System Three-Quarter-Inch Line Break During Surveillance Testing While Plant was Shutdown - Component Failure." This LER described an event where, during surveillance testing, a 3/4-inch high point vent line failed approximately one-half inch above a fillet weld due to fatigue. Corrective actions included 1) repairing the high point vent line piping, and 2) implementation of a design modification to reduce vibration on the HPCS Suppression Pool Test Return Line (this is currently scheduled for the 1992 maintenance and refueling outage).

3. LER 90-028, "Degradation of Primary Containment Pressure Boundary Caused by a Plant Shutdown Due to Cracks on High Pressure Core Spray Small Bore Piping." This LER described three separate reportable events pertaining to cracks and linear indications that were discovered in the HPCS System. Corrective actions included 1) performing an extensive evaluation of HPCS small bore piping welds which had previously been identified as having some probability of failure due to HPCS System vibration and pipe configuration, and 2) repairing two HPCS vent lines and one HPCS drain line.

NRC FORM 366A
(6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

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

EIIS Information

Text Reference	EIIS Reference System	Component
Reactor Drywell	NH	---
Main Condenser	SG	COND
Residual Heat Removal (RHR) System	BO	---
RHR-V-161A	BO	V
RHR-V-112A	BO	ISV
Reactor Recirculation (RRC) System	AD	---
High Pressure Core Spray (HPCS) System	BG	---
RHR-V-37/38	BO	V
RHR-V-157B/158B	BO	V
RHR-V-163/164	BO	V
RHR-V-113	BO	V
Leak Detection System	LJ	---