

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8907190359 DOC. DATE: 89/07/14 NOTARIZED: NO DOCKET #
 FACIL: 50-397. WPPSS. Nuclear Project, Unit 2, Washington Public Power 05000397
 AUTH. NAME AUTHOR AFFILIATION
 DAVISION, W.S. Washington Public Power Supply System
 POWERS, C.M. Washington Public Power Supply System
 RECIP. NAME RECIPIENT AFFILIATION

SUBJECT: LER 89-024-00: on 890614, secondary containment bypass leakage
 found greater than allowed by design basis.

W/8 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
 TITLE: 50.73/50.9 Licensee Event Report (LER), Incident Rpt, etc.

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR ENCL
	PD5 LA	1 1	PD5 PD	1 1
	SAMWORTH, R	1 1		
INTERNAL:	ACRS MICHELSON	1 1	ACRS MOELLER	2 2
	ACRS WYLIE	1 1	AEOD/DOA	1 1
	AEOD/DSP/TPAB	1 1	AEOD/ROAB/DSP	2 2
	DEDRO	1 1	IRM/DCTS/DAB	1 1
	NRR/DEST/ADE 8H	1 1	NRR/DEST/ADS 7E	1 0
	NRR/DEST/CEB 8H	1 1	NRR/DEST/ESB 8D	1 1
	NRR/DEST/ICSB 7	1 1	NRR/DEST/MEB 9H	1 1
	NRR/DEST/MTB 9H	1 1	NRR/DEST/PSB 8D	1 1
	NRR/DEST/RSB 8E	1 1	NRR/DEST/SGB 8D	1 1
	NRR/DLPQ/HFB 10	1 1	NRR/DLPQ/PEB 10	1 1
	NRR/DOEA/EAB 11	1 1	NRR/DREP/RPB 10	2 2
	NUDOCS ABSTRACT	1 1	REG FILE 02	1 1
	RES/DSIR/EIB	1 1	RES/DSR/PRAB	1 1
	RGN5 FILE 01	1 1		
EXTERNAL:	EG&G WILLIAMS, S	4 4	FORD BLDG HOY, A	1 1
	L ST LOBBY WARD	1 1	LPDR	1 1
	NRC PDR	1 1	NSIC MAYS, G	1 1
	NSIC MURPHY, G. A	1 1		

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK,
 ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION
 LISTS FOR DOCUMENTS YOU DON'T NEED!

FULL TEXT CONVERSION REQUIRED
 TOTAL NUMBER OF COPIES REQUIRED: LTTR 43 ENCL 42

R
I
D
S
/
A
D
D
S
/
A
D
D
S

1104 gh

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

Docket No. 50-397

July 14, 1989

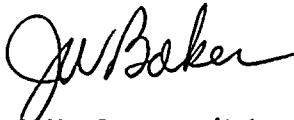
Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 89-024

Dear Sir:

Transmitted herewith is Licensee Event Report No. 89-024 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,



G.M. Powers (M/D 927M)
WNP-2 Plant Manager

CMP:lg

Enclosure:
Licensee Event Report No. 89-024

cc: Mr. John B. Martin, NRC - Region V
Mr. C.J. Bosted, NRC Site (M/D 901A)
INPO Records Center - Atlanta, GA
Ms. Dottie Sherman, ANI
Mr. D.L. Williams, BPA (M/D 399)

8907190359 890714
PDR ADOCK 05000397.
S PDC

Cert No
PN14192724
TER2
11

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) Washington Nuclear Plant - Unit 2															DOCKET NUMBER (2) 0 5 0 0 0 3 9 1 7 1 OF 0 1 6										PAGE (3)				
TITLE (4) Secondary Containment Bypass Leakage Found To Be Greater Than Allowed By Design Basis as a Result of Equipment Design Deficiency																													
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)					
0 6 1 4			8 9		8 9		0 2 4		0 0		0 7 1 4 8 9							0 5 0 0 0											
OPERATING MODE (9) 5					THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)																								
POWER LEVEL (10) 0 0 0					20.402(b)					20.405(c)					<input checked="" type="checkbox"/> 50.73(a)(2)(iv)					73.71(b)									
					20.405(a)(1)(i)					50.38(c)(1)					50.73(a)(2)(v)					73.71(c)									
					20.405(a)(1)(ii)					50.38(c)(2)					50.73(a)(2)(vii)					OTHER (Specify in Abstract below and in Text, NRC Form 366A)									
					20.405(a)(1)(iii)					50.73(a)(2)(i)					50.73(a)(2)(viii)(A)														
					20.405(a)(1)(iv)					<input checked="" type="checkbox"/> 50.73(a)(2)(ii)					50.73(a)(2)(vii)(B)														
					20.405(a)(1)(v)					50.73(a)(2)(iii)					50.73(a)(2)(ix)														
LICENSEE CONTACT FOR THIS LER (12)																													
NAME W.S. Davison, Compliance Engineer															TELEPHONE NUMBER AREA CODE 510 1 9 317 1 71-12 1 712 16														
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)																													
CAUSE		SYSTEM		COMPONENT		MANUFACTURER		REPORTABLE TO NPDs				CAUSE		SYSTEM		COMPONENT		MANUFACTURER		REPORTABLE TO NPDs									
SUPPLEMENTAL REPORT EXPECTED (14)															EXPECTED SUBMISSION DATE (15)					MONTH DAY YEAR									
<input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE)															<input checked="" type="checkbox"/> NO														
ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)																													
<p>On June 14, 1989, during a plant design review for unmonitored containment release paths, a Plant Design Engineer identified a potential path through which radioactive liquid from the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. The bypass leakage path identified was the Control Rod Drive (CRD) System Hydraulic Control Unit (HCU) control valves. This postulated event was evaluated as having the potential to result in bypass leakage of a quantity large enough to violate the WNP-2 design basis for control room habitability radiation dose limits after a Loss Of Coolant Accident (LOCA). An engineering assessment determined that installation of two check valves in series in the common discharge line of the CRD pumps would provide a barrier sufficient to prevent bypass leakage through the CRD system. These valves were installed on June 24, 1989, prior to the end of the refueling outage before the plant startup. The immediate cause of this event was equipment design deficiency in that the CRD system design was potentially not capable of preventing post LOCA liquid bypass leakage from exceeding the design basis limit. The plant design review to verify that there are no other potential unmonitored release paths will continue as originally described in LER 88-012-00. The probability of occurrence of a LOCA event of sufficient magnitude to cause major fuel damage, concurrent with a seismic event causing failure of the condensate system outside of secondary containment is very low. Since no event actually occurred at WNP-2 during the time prior to correction of the CRD system bypass leakage problem, this event posed no threat to the health and safety of Plant personnel or the public.</p>																													

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	8 9	— 0 2 4	— 0 0	0 2	OF	0 5

TEXT (If more space is required, use additional NRC Form 368A's) (17)

Plant Conditions

- a) Plant Mode - 5 (Refueling)
- b) Power Level - 0%

Event Description

On June 14, 1989, during a plant design review for unmonitored containment release paths, a Plant Design Engineer identified a potential path through which radioactive liquid from the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. The bypass leakage path identified was the Control Rod Drive (CRD) System Hydraulic Control Unit (HCU) control valves. These valves potentially were capable of allowing backleakage from the reactor vessel through 740 operating lines through the CRD pumps and into the Condensate System which is located outside of the secondary containment. This postulated event was evaluated as having the potential to result in bypass leakage of a quantity large enough to violate the WNP-2 design basis for control room habitability radiation dose limits after a Loss Of Coolant Accident (LOCA). The evaluation was being performed as part of corrective action for LER 88-012-00 "Potential Unmonitored Effluent Release Path Due to Design Error by Architect/Engineer Cause Unknown".

The CRD System connects the Condensate System to the reactor vessel via the suction line for the system pumps CRD-P-1A and CRD-P-1B. The system pump common discharge line is connected to 185 Hydraulic Control Units (HCU) by 370 two inch and 370 one inch CRD system operating lines. Each HCU is connected to a CRD Mechanism (attached to the reactor vessel bottom head) by a one inch control rod insert operating line and a three quarter inch control rod withdraw operating line. This design configuration results in 740 parallel leak paths which, after a LOCA, potentially could allow water to flow from the reactor vessel through the CRD System and into the Condensate System thus bypassing the secondary containment system.

Immediate Corrective Action

An engineering assessment determined that installation of two check valves in series in the common discharge line of the CRD pumps would provide a barrier sufficient to prevent bypass leakage through the CRD system. These valves were installed on June 24, 1989, and leak tested to confirm compliance with leakage criteria prior to the end of the refueling outage, before the plant startup

Further Evaluation and Corrective ActionA. Further Evaluation

1. This event is reportable per the following:

- o 10CFR50.73(a)(2)(ii)(B)- as a condition that was outside the design basis of the plant.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	8 9	— 0 2 4	— 0 0	0 3	OF	0 5

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

- o 10CFR50.73(a)(2)(v)- as a condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to: (C) control the release of radioactive material; or (D) mitigate the consequences of an accident.
2. There were no plant structures, components, or systems inoperable at the start of this event that contributed to the event.
3. The immediate cause of this event was equipment design deficiency in that the CRD system design was potentially not capable of preventing post LOCA liquid bypass leakage from exceeding the design basis limit. The root cause investigation is still in progress. Any substantive information resulting from its completion will be reported in a supplementary LER.
4. The total potential liquid bypass leakage from the CRD System prior to installation of the two check valves was estimated by assuming leakage based on the ASME Section XI leakage criteria as follows:

<u>Valve Designation</u>	<u>Number Of Valves Installed</u>	<u>Leakage Per Valve</u>
CRD-V-126XXXX CRD-V-115XXXX	185	0.018 GPM
CRD-V-120XXXX CRD-V-123XXXX	185	0.018 GPM
CRD-V-122XXXX CRD-V-121XXXX	185	0.018 GPM
CRD-V-138XXXX	185	0.036 GPM
TOTAL LEAKAGE		16.65 GPM = 134.0 SCFH

The design basis total allowable bypass leakage rate is 0.74 SCFH

XXXX= CRD NUMBER DESIGNATION-EACH CRD HAS ONE HCU CONTAINING 7 VALVES WHICH ISOLATE FOUR OPERATING LINES FOR A TOTAL OF 1295 VALVES WHICH ISOLATE 740 INDIVIDUAL LINES.

CRD-V-126XXXX = The inlet scram valve for each HCU

CRD-V-115XXXX = Check valve installed on the pump side of the CRD-V-126XXXX valve for each HCU.

CRD-V-120XXXX = Over piston directional control valve for each HCU.

CRD-V-123XXXX = Over piston directional control valve for each HCU.

CRD-V-122XXXX = Under piston directional control valve for each HCU.

CRD-V-121XXXX = Under piston directional control valve for each HCU.

CRD-V-128XXXX = Check valve installed on the cooling water line for each HCU.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (8)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
Washington Nuclear Plant - Unit 2	0 5 0 0 0 3 9 7	8 9	— 0 2 4	— 0 0	0 4	OF	0 5

TEXT (If more space is required, use additional NRC Form 368A's) (17)

B. Further Corrective Action

1. The plant design review to verify that there are no other potential unmonitored release paths requiring monitoring will continue as originally described in LER 88-012-00.
2. General Electric, Inc., the WNP-2 Nuclear Steam Supply System contractor, has been notified of this condition at WNP-2. This condition is not being reported to the NRC as a 10CFR21 violation because, as specified in 10CFR21.1, the Supply System has knowledge that the Commission has been adequately informed of such defects. Correction of this design deficiency was included in granting the Susquehanna and LaSalle plant operating licenses.

Safety Significance

The maximum potential release of liquid and gaseous radioactive material via bypass leakage paths for WNP-2 is analyzed as 0.74 scfh. This will ensure that control room dose and radioactive release to the environment is within the range allowed by 10CFR100. The potential for leakage as a result of the newly identified CRD system bypass path was analyzed as 134 scfh. This is many times the allowable limit and could potentially have resulted in the release of significantly more radioactive material than predicted by design basis calculations under the worst-case post LOCA assumptions. The probability of occurrence of a LOCA event of sufficient magnitude to cause major fuel damage, concurrent with a seismic event causing failure of the condensate system outside of secondary containment is very low. Since no event actually occurred at WNP-2 during the time prior to correction of the CRD system bypass leakage problem, this event posed no threat to the health and safety of Plant personnel or the public.

Similar Events

LER 88-012-00 documented a condition in which a potential unmonitored release path for radioactive material was identified which might allow air flow from the Turbine Building through the Diesel Generator Building corridor to the atmosphere. This condition was also caused by equipment design deficiency. This LER (89-024-00) is a direct result of corrective action formulated during the event documented by LER 88-012-00.

EIIS Information**Text Reference****EIIS Reference**

System	Component
--------	-----------

Primary Containment	NH	---
Secondary containment	NG	---
Control Rod Drive System	AA	---

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104

EXPIRES: 8/31/88

FACILITY NAME (1) Washington Nuclear Plant - Unit 2	DOCKET NUMBER (2) 0 5 0 0 0 3 9 7	LER NUMBER (6)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		8 9	0 2 4	0 0	0 5	OF	0 5

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

IIIS InformationText ReferenceIIIS Reference

System Component

Hydraulic Control Unit

AA HCU

Control Valves

AA XCV

Reactor Vessel

AC RPV

Condensate System

SD ---

CRD-P-1A, CRD-P-1B

AA P

CRD Mechanism

AA 84

Check Valves

AA V

CRD-V-126XXX, -115XXX, -120XXX, -123XXX, -122XXX,
-121XXX, -138XXX

AA V

Turbine Building

NM ---

Diesel Generator Building

VJ ---