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POWERS,C.M. Washington Public Power Supply System
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SUBJECT: LER 89-024-00:on 890614,secondary containment bypass leakage found greater than allowed by design basis.
W/8 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

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,

JW Baker

C.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)

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NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

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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
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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)	[ ] 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)	[ ] 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	TELEPHONE NUMBER	
	AREA CODE	
W.S. Davison, Compliance Engineer	5 0 9	3 7 7 - 2 7 2 6

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)

[ ] YES *(If yes, complete EXPECTED SUBMISSION DATE)* [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)* (16)

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 prvent 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.

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TEXT (If more space is required, use additional NRC Form 366A'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 oeprating lines. Each HCU is connected to a CRD Mechanism (attached to the reactor vessl 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 Condesnate 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 Action

A. 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.

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- 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:

Valve Designation	Number Of Valves Installed	Leakage Per Valve
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.

*U.S. GPO: 1988-520-589/00070

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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 condesate 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	---

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EIIS Information

Text Reference	EIIS 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-126XXXX,-115XXXX,-120XXXX,-123XXXX,-122XXXX, -121XXXX,-138XXXX	AA	V
Turbine Building	NM	---
Diesel Generator Building	VJ	---

*U.S. GPO: 1988-520-589/00070