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SUBJECT: LER 89-027-00:on 890630,inadequate seismic restraint of isolation valves could result in unisolatable breach of PC. W/8 ltr.

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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 28, 1989

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

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 89-027

Dear Sir:

Transmitted herewith is Licensee Event Report No. 89-027 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,

CMPowers

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

CMP:lg

Enclosure:
Licensee Event Report No. 89-027

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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4) INADEQUATE SEISMIC RESTRAINT OF ISOLATION VALVES COULD RESULT IN UNISOLATABLE BREACH OF PRIMARY CONTAINMENT CAUSED BY INADEQUATE WORK PRACTICES

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	3 0	8 9	8 9	0 2 7	0 0	0 7	2 8	8 9		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 2 5

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)	[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)	[ ] 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)
R.E. Fuller, Compliance Engineer	5 0 9 3 7 7 - 2 7 9 7

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)

[ ] 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 30, 1989 a preliminary engineering evaluation determined that two seismic supports missing on each of two Post Accident Sampling System (PASS) containment isolation valves, found by a Design Engineer on June 27, 1989, would probably result in failure of the pipe at its Primary Containment penetration during a Design Basis Earthquake (DBE). This would create an unisolatable breach of Primary Containment. The "as found" condition was discovered while the Design Engineer was performing a visual inspection of plant supports and while the Plant was at 3% power and in Mode 2 (Startup).

At 1650 hours on June 30, 1989 the Primary Containment Technical Specification action statement 3.6.1.1 was entered and preparations were made to restore restraints to the required Plant configuration. At 1745 hours when work was not completed on the restraints, a Plant shutdown was initiated. Primary Containment Technical Specification action statement was exited at 1843 hours when the restraints were restored.

The root causes of the event are 1) less than adequate work practices to ensure the Plant configuration remains within design requirements, and 2) less than adequate training of project personnel to implement Plant modifications.

No further corrective actions were identified that would significantly minimize the recurrence of this condition in future Plant modifications. Current programs and procedures are considered adequate to ensure the plant configuration remains within the design requirements.

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

Abstract (cont'd)

Based on engineering judgement, there is no safety significance associated with this event because a qualitative assessment determined that a more rigorous stress analysis would indicate the pipe not fail from a Design Basis Earthquake (DBE). Since the condition did not actually occur, this condition did not threaten the health and safety of the public and Plant personnel.

Plant Conditions

a) Power Level - 25%
b) Plant Mode - 1 (Power Operation)

Event Description

On June 30, 1989 a preliminary engineering evaluation determined that two seismic supports missing on each of two Post Accident Sampling System (PASS) containment isolation valves, found by a Design Engineer on June 27, 1989, would probably result in failure of the pipe at its Primary Containment penetration during a Design Basis Earthquake (DBE). This would create an unisolatable breach of Primary Containment. The "as found" condition was discovered while the Design Engineer was performing a visual inspection of plant supports and while the Plant was at 3% power and in Mode 2 (Startup).

The two containment isolation valves (PSR-V-X82/1 & PSR-V-X82/2) are installed in series on a 1-inch stainless steel penetration line (PI(1)-4S-X82d) leading to the Suppression Pool. The valves are located in the Reactor Building just outside of Primary Containment.

For valve PSR-V-X82/1, two angle iron braces, which provide vertical seismic restraint, were missing from the support. Two U-bolts were missing from the valve PSR-V-X82/2 support. The U-bolts provide three directional seismic restraint.

A preliminary engineering evaluation was performed on the 1-inch PASS line without the seismic supports on the two PASS valves. The evaluation conservatively used elastic modeling techniques, which does not allow for plastic deformation and results in higher stress values. The evaluation determined the highest stress in the pipe would occur at the penetration to Primary Containment and would be sufficient to fail the pipe.

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

Immediate Corrective Action

At 1650 hours on June 30, 1989 the Primary Containment Technical Specification action statement 3.6.1.1 was entered and preparations made to restore restraints to the required Plant configuration. At 1745 hours when work was not completed on the restraints, a Plant shutdown was initiated. The U-bolts were installed on PSR-V-X82/2 and the Primary Containment Technical Specification action statement exited at 1843 hours. An engineering evaluation had determined that the U-bolts would provide adequate seismic restraint to prevent failure of the pipe during a DBE. The angle iron braces for PSR-V-X82/1 were installed a short time later. A Plant shutdown had been planned for the evening of July 30, 1989 and the Plant was shutdown by a planned manual scram at 2323 hours.

Further Evaluation and Corrective Action

A. Further Evaluation

1. This event is reportable per 10 CFR 50.73(a)(2)(ii)(B) and 10 CFR 50.73(a)(2)(v)(C) & (D) as a condition that was outside of the design basis of the plant and alone could have prevented the fulfillment of the safety function of Primary Containment to control the release of radioactive material and mitigate the consequences of an accident. Preliminary engineering analyses determined the configuration could result in pipe failure at the containment penetration location following a Design Basis Earthquake (DBE), causing a breach of Primary Containment.

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

3. An ASME Section XI plan (Plan No. 2-0112) was issued November 1, 1983 to reverse the flow direction of the two PASS valves. This required each valve to be removed from the line and reinstalled in reverse direction to provide a more reliable pressure seal on the Primary Containment side of the valve. The Section XI plan did not require supports or restraints to either be removed or reinstalled during performance of this task.

   There is no documentation to indicate modification occurred later on these valves and/or associated lines that required removal of the restraints. Therefore, it is assumed that the two PASS valves were left in the above described configuration following implementation of the Section XI plan 2-0112.

4. The root causes of the event are 1) less than adequate work practices to ensure the Plant configuration remains within design requirements, and 2) less than adequate training of project personnel to implement Plant modifications.

a) A change notice was implemented to remove and reinstall supports for valve PSR-V-X82/7 on line PI(1)-4S-X82f, similar to work being performed on PSR-V-X82/1 at the same time. However, change notice documentation was not provided for the supports on PSR-V-X82/1 AND PSR-V-X82/2. The work practices were less than adequate to coordinate identification of Section XI plan deficiencies and to perform adequate post-modification inspections.

b) Project personnel lacked training to ensure Plant modifications were within the required design configuration. Either the personnel responsible for modification of line PI(1)-4S-X82d did not recognize removal of the supports was not authorized by the Section XI plan, and/or they were not aware that a change notice was required for work that was not specifically identified in the Section XI plan. In either case, project personnel were less than adequately trained to ensure that Plant modifications are clearly documented and approved to ensure compliance with design requirements.

5. Programs and procedure revisions have been implemented since the event occurrence (and not as a result of the occurrence) to provide added assurance that the Plant configuration remains within the required design configuration. As a result of the Safety System Functional Inspection (SSFI) and subsequent to 1987, each Plant Technical System Engineer is required to perform a visual inspection of their assigned system prior to Plant startup from a major outage. The condition reported herein was missed during two previous inspections because it is located in a hard to reach area and it was known that that part of the system had had no recent major modifications. This condition was discovered while in the course of responding to a general Plant Management directive to all engineers to perform random inspections of the Plant in areas of their expertise for conformance to the required Plant configuration.

In addition, the Plant Modification Request (PMR) procedure (PPM 1.4.1) was revised to require a post-modification review or inspection by the Design Engineer and the Plant Technical System Engineer of selected Plant modifications based upon a selection criteria. Also, the Plant procedure PPM 1.3.19 has been revised to require Area Coordinators to be trained to identify degradation and abnormalities of equipment. Furthermore, the Project Engineer of a Plant modification is responsible for ensuring that the Plant configuration remains within the design requirements.

B. Further Corrective Action

No further corrective actions were identified that would significantly minimize the recurrence of this condition in future Plant modifications. However, current Plant programs and procedures are constantly being reviewed to identify areas where improvements can be made to provide increased confidence that the Plant configuration will remain within the design requirements. Also, current programs provide for continual review of the existing Plant configuration for compliance with the design requirements.

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

Safety Significance

Based on engineering judgement, there is no safety significance associated with this event. A qualitative assessment determined that a more rigorous stress analysis with sophisticated plastic modeling techniques would indicate the PASS line (PI(1)-4S-X82d) would not fail from a Design Basis Earthquake (DBE). However, since the preliminary engineering analysis determined the pipe would fail, the safety significance of the postulated event (DBA, Earthquake and LOCA) is indeterminate because the effect of radionuclide release to the Reactor Building has not been analyzed.

Since the condition did not actually occur, this condition did not threaten the health and safety of the public and Plant personnel.

Similar Events

None

EIIS Information

Text Reference	EIIS Reference System	Component
Sampling and Water Quality System	KN	---
Sampling and Water Quality System (PSR-V-X82/1)	KN	ISV
Sampling and Water Quality System (PSR-V-X82/2)	KN	ISV
Reactor Building	NG	---
Reactor Containment	NH	---