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SUBJECT: LER 90-029-00:on 901108,inadequate containment penetration testing associated w/wetwell hatch pressure warning devices.

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

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Docket No. 50-397

December 7, 1990

G02-90-203

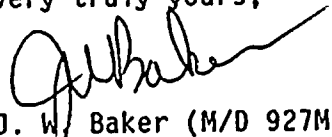
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Washington, D.C. 20555

Subject: NUCLEAR PLANT NO. 2
LICENSEE EVENT REPORT NO. 90-029

Dear Sir:

Transmitted herewith is Licensee Event Report No. 90-029 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,


J. W. Baker (M/D 927M)
WNP-2 Plant Manager

JWB:lr

Enclosure:
Licensee Event Report No. 90-029

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

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LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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 1 OF 0 4										PAGE (3) 1 OF 0 4				
TITLE (4) Inadequate Containment Penetration Testing Associated With Wetwell Hatch Pressure Warning Devices																								
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	8	9	0	9	0	1	2	0	7	9	0	0 5 0 0 0					0 5 0 0 0					
OPERATING MODE (9)		4		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)																				
POWER LEVEL (10)		0		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)(ix)												
LICENSEE CONTACT FOR THIS LER (12)																								
NAME C. L. Fies, Compliance Engineer										TELEPHONE NUMBER 5 0 1 9 3 1 7 1 - 1 2 0 1 3 9														
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														
B	N/H	P/E/N	Ti 3 3 0	N																				
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 8, 1990, during final preparations for startup following a nine day forced outage, four previously unidentified containment isolation boundaries were discovered on the Primary Containment. On two of these isolation boundaries (those associated with the Containment Wetwell Hatch (C-X-51)) the 10CFR50, Appendix J local leak rate tests have not been performed in accordance with the Technical Specification Surveillance Requirements of Paragraph 4.6.1.1.a. This is a reportable condition under 10CFR50.73(a)(2)(i)(B). This condition was discovered when one of the boundaries was broken during installation of the hatch. The other two isolation boundaries have not been disturbed between (or since) Integrated Leak Rate Tests (ILRTs) and, therefore, do not require local leak rate testing.

Immediate corrective action was taken to repair and leak test the containment isolation boundaries.

The root cause of this event was inadequate communication of this design feature from the Vendor/Architect Engineer to the Supply System. A contributing cause was Leak Test Procedures less than adequate as they did not cover the situation.

The event posed no threat to the health and safety of either the public or plant personnel.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH 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.

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

Plant Conditions

Power Level - 0%

Plant Mode - 4

Event Description

At approximately 2030 hours on November 8, 1990 Plant Maintenance personnel reported a broken part on the Containment Wetwell Hatch (C-X-51) closure. The WNP-2 Containment Wetwell hatch is a Yoke-type closure approximately 42 inches in diameter that provides access to the wetwell from the Reactor Building. At the time, the plant was in the final preparations for startup after a nine day forced outage (see LER 90-028). The closure of the wetwell is a routine operation that is performed after each entry during refueling outages and forced outages that require wetwell access.

Inspection of the broken part by the Plant Maintenance Engineer along with a review of vendor documentation showed the part to be a portion of a pressure warning device provided by a Containment subvendor. The pressure warning device consists of a nipple, gasket and holding screw (See attached Figure A, Items 1, 2, and 3). The pressure warning device is mounted on the head of the closure (See attached Figure B, Item 27) and is designed to provide an audible indication of high pressure in the wetwell before the yoke closure is opened. The nipple associated with the pressure warning device is welded to the Containment Wetwell Hatch Head at the point where a one-quarter inch hole penetrates the hatch. The containment boundary is then the gasketed joint between the nipple and the holding screw.

Up until the time the nipple broke, the Plant Staff was not aware of this gasket boundary associated with the Wetwell Hatch Head. The nipple was thought to be a stud welded to the outside of the head. Consequently the 10CFR50, Appendix J leak rate tests have not been performed on this gasketed boundary joint. This is not in accordance with the Technical Specification Surveillance Requirements of Paragraph 4.6.1.1.a and is therefore a reportable event.

Immediate Corrective Action

A Plant Operating Committee (POC) Immediate Disposition Approval Request was approved which authorized the grind out of the old seal weld for the nipple. An ASME Section XI Repair Plan was developed to fabricate and install the replacement nipple.

LICENSEE EVENT REPORT (LER)
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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
		9 0	0 2 9	0 0	0 3	OF 0	4

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

Further Evaluation and Corrective ActionA. Further Evaluation

1. This event is being reported per the requirements of 10CFR50.73(a)(2)(i)(B) as "Any operation or condition prohibited by the plant's Technical Specifications;" Specifically, the requirement for leak rate testing Containment isolation boundaries was not being met.
2. Plant Mechanics did not have a torque value available for the holding screw attached to the nipple. It is believed the holding screw was over torqued when the wetwell hatch was being installed after the forced outage.
3. Further evaluation showed that the design provided no means of performing a leak rate test of the pressure warning device.
4. Inspection of the second (bottom) Pressure Warning device showed a crack had developed in the nipple on this device also. Consequently, the repair plan was modified to replace both nipples prior to startup.
5. The root cause of this event was inadequate communication of this design feature from the Vendor/Architect Engineer to the Supply System. A contributing cause was Leak Test Procedures less than adequate as they did not cover the situation.
6. Similar pressure warning devices (two) are located on the Control Rod Drive (CRD) Access Hatch which is part of the Equipment Hatch. However, the CRD Access Hatch has not been used since the performance of the Integrated Leak Rate Test (ILRT) and, therefore, does not require a leak rate test.
7. There were no structures, components or systems that were inoperable prior to the start of this event which contributed to the event.

B. Further Corrective Action

1. Surveillance procedure PPM 7.4.6.1.1.1, Pressure Decay Leak Rate Testing, was utilized with specially built tooling to provide a method of testing the Pressure Warning devices.
2. A caution tag was placed on the hatch giving the correct torque values for the holding screw.
3. The pressure warning devices will be replaced by solid studs to eliminate these unnecessary containment isolation boundaries.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

Safety Significance

The pressure warning devices have been tested as part of the Containment Integrated Leak Rate Test (ILRT). This test was run during plant startup and during the second refueling outage. During these tests there was no known indication of any leakage through these penetrations.

The safety significance of this event is also bounded by the instrument line break accident analysis described in Section 15.6.2 of the FSAR. This accident analysis describes a break in an instrument line which contains a one-quarter inch diameter flow orifice. The analysis assumes the line is connected to the primary coolant system, is not isolatable, and the break occurs in the Secondary Containment. This would be more severe than the one-quarter inch opening in the wetwell hatch since it is not connected directly to the primary containment. The calculated radiological effects of the instrument line break analysis are several orders of magnitude below 10CFR100 limits.

Similar Events

There are no similar events.

EIIS InformationText ReferenceEIIS Reference

	<u>System</u>	<u>Component</u>
Primary Containment	NH	--
Containment Wetwell	BT	--
Containment Wetwell Hatch (C-X-51)	BT	PEN
Reactor Building	NG	--
Secondary Containment	NG	--