NRC Form 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

(See reverse for number of digits/characters for each block)

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TITLE (4)
Plant Valve Configuration for inerting Primary Containment was outside the Design Basis of the Plant

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 NAME N/A	DOCKET NUMBER 05000
10	23	97	97	024	00	11	24	97	FACILITY NAME N/A	DOCKET NUMBER 05000

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)							
			20.2201 (b)		20.2203(a)(2)(v)		50.73(a)(2)(i) (B)		50.73(a)(2)(viii)
POWER LEVEL (10)	100		22.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)(B		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v) (D)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert L. Cannon - Senior Regulatory Affairs Engineer	508-830-8321

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

On October 23, 1997, during a review of the primary containment atmosphere control system as a result of a General Electric (GE) notification of a potential 10CFR21 issue regarding suppression pool (torus) bypass leakage, one potential bypass flow path was identified through which the drywell air space could communicate with the torus air space. The potential for this event existed while in the valve line up for inerting the primary containment because a single failure could occur that causes valves AO-5035A and PCIS AO-5036A to fail open on a primary containment isolation signal. These valves share a common actuating relay, RPWA1 and relay contact 1-2, that provide the primary containment isolation system signal to these valves. If the plant was in the process of changing from torus to drywell inerting and a LOCA event occurred, a single failure of relay RPWA1 to deenergize or the single failure of relay RPWA1 contact 1-2, to open would prevent AO-5035A and AO-5036A from closing, resulting in a steam bypass path from the drywell to the torus air space without passing through the torus downcomers. This operating configuration was determined to be outside the design basis of the of the plant. Immediate action was taken to establish administrative controls that preclude entry into the inerting portion of Pilgrim Nuclear Power Station Procedure 2.2.70, "Primary Containment Atmosphere Control System." Procedure 2.2.70, will be revised to eliminate this vulnerability. This event posed no significant threat to the public health and safety.

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

EVENT DESCRIPTION

On October 23, 1997, during a review of the primary containment atmosphere control system as a result of a General Electric's (GE) notification of a potential 10CFR21 issue regarding suppression pool (torus) bypass leakage, one potential bypass flow path was identified through which the drywell air space could communicate with the torus air space while in the process of changing from torus inerting to drywell inerting if a single failure occurred that caused valves AO-5035A and AO-5036A to fail to close on a PCIS signal. These valves share a common primary containment isolation system (PCIS) actuating relay, RPWA1 and relay contact 1-2. If the plant was in the process of changing from torus to drywell inerting and a LOCA event occurred, a single failure of relay RPWA1 to deenergize or the single failure of relay RPWA1 contact 1-2 to open would prevent AO-5035A and AO-5036A from automatically closing on a PCIS signal, resulting in a steam bypass path from the drywell to the torus without passing through the torus downcomers. This vulnerability was evaluated to be outside the design basis of the of the plant.

On October 23, 1997, Problem Report 97.9652 (corrective action program document) was prepared by the operations department manager to document the identified problem.

At the time of discovery, the plant wa operating at approximately 100 percent power with the mode switch in the RUN position. The reactor vessel pressure was at approximately 1035 psig with the reactor water at the saturation temperature for vessel pressure. This event posed no significant threat to public health and safety.

CAUSE

When Procedure 2.2.70, "Primary Containment Atmosphere Control System" was developed, it was not recognized that the procedure-specified valve lineup for inerting the primary containment could result in a bypass flow path from the drywell atmosphere to the torus atmosphere given a single failure of a PCIS relay. During a LOCA event, if relay RPWA1 failed to deenergize or if relay RPWA1 contact 1-2 failed to open on a containment isolation signal, valves AO-5035A and AO-5036A would remain open and create a bypass path from the drywell to the torus air space via manual isolation valve 9-HO-117.

IMMEDIATE CORRECTIVE ACTION

Immediate action was taken to initiate a tracking Limiting Condition for Operation (LCO) not allowing entry into the inerting portion of Procedure 2.2.70 until corrective actions have been completed.

CORRECTIVE ACTION TO PRECLUDE RECURRENCE

Procedure 2.2.70 will be revised to eliminate the single failure vulnerability that would result in a bypass path from the drywell to the torus air space during the swap from torus to drywell inerting.

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SAFETY SIGNIFICANCE

This event posed no significant threat to the public health and safety.

Though the pressure suppression function was not operable during the short time period this configuration existed, the probability of the combination of events leading to the bypass path during a LOCA is vanishingly small.

In order to result in conditions beyond the analyzed envelope, a LOCA must occur and PCIS relay RPWA1 must fail to deenergize or its 1-2 contact fail to open, and AO5035A, AO5036A, and 9-HO-117 must be open. This specific valve lineup occurs only briefly as part of the sequence that shifts from tours inerting to drywell inerting.

The PNPS IPE report lists the probabilities for all LOCA events at 1.17E-2 per year. The probability for a failure of an AC or DC control relay, according to IEEE 500, is 4.54E-6 per demand.

The window during which AO5035A, AO5036A, and 9-HO-117 are all open is only during containment inerting operations and only during the performance of step 7.1[21] in PNPS Procedure 2.2.70, Revision 56. This step involves six actions, all of which are performed sequentially at panel C7 in the control room by a single operator. Within this step, there are only two actions between the opening of AO-5035A and closing AO-5036A, and these actions are closing the two torus purge exhaust valves.

Assuming that the drywell inerting procedure is performed once per year, and assuming that the two valves are opened for 5 minutes during this step, the probability per year that these valves are open is conservatively:

5 minutes / 5.26E5 minutes per year = 9.51E-6 per year.

Therefore, the probability that RPWA1 will fail in response to a LOCA during the time when all three valves are opened is:

Probability of LOCA * Per demand failure of RPWA1 * Probability of valves open

1.17E-2 * 4.54E-6 * 9.51E-6 = 5E-13

Additionally, as described in the Bases of Technical Specification 3.5.A, the NRC recognizes the probability of occurrence of a LOCA during the 24 hour window following reactor startup is very low.

This report is submitted in accordance with 10 CFR 50.73(a)(ii)(B) because the single failure vulnerability was outside the plant design basis. On November 20, 1997, at 1510 hours a 10CFR50.72(b)(1)(ii)(B) telephone notification was made to the NRC Operations Center. Problem Report 97.9710 was written to document the notification.

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SIMILARITY TO PREVIOUS EVENTS

A review of Licensee Event Reports (LERs) issued since 1984 was conducted. The review focused on reports submitted in accordance with 10 CFR 50.73(a)(ii)(B) which involved single failure vulnerability. The review identified a related instance reported in LER 97-011-01

LER 97-011-01 reported the Pilgrim station design basis includes a requirement for redundant and independent salt service water (SSW) system trains such that no single active failure can prevent the SSW system from fulfilling its safety objective (i.e., to provide cooling water to the reactor building closed cooling water (RBCCW) system. The design basis of the SSW system also includes the requirements for the normally cross-connected SSW trains to be automatically isolated upon loss of the preferred AC power source. During a Service Water Operational Performance Inspection follow-up NRC inspection, a single failure vulnerability was identified which placed the unit in a condition thought to be outside the design basis. Specifically, a single failure of a 125 vdc battery, under certain conditions, would compromise the redundancy and independence of the SSW system and potentially lead to a SSW pump cavitation condition. This LER is targeted for near term update.

ENERGY INDUSTRY IDENTIFICATION (EIIS) CODES

COMPONENTS	CODES
Relay	RLY
Valve	V

SYSTEMS

Primary Containment Atmosphere Control System	BB