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SUBJECT: LER 93-013-00:on 930318,design error identified by engineers working in component safety calssification program.Caused by deficiency in original design of CIS.RCIC pump isolation valve from suppression de-energized.W/930419 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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April 19, 1993
G02-93-088

Docket No. 50-397

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

Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21**
LICENSEE EVENT REPORT NO. 93-013

Transmitted herewith is Licensee Event Report No. 93-013 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.

Sincerely,

JW Baker

J. W. Baker
WNP-2 Plant Manager (Mail Drop 927M)

JWB/CDM/cgeh
Enclosure

cc: Mr. J. B. Martin, NRC - Region V
Mr. R. Barr, NRC Resident Inspector (Mail Drop 901A, 2 Copies)
INPO Records Center - Atlanta, GA
Mr. D. L. Williams, BPA (Mail Drop 399)

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LICENSEE EVENT REPORT (LER)

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TITLE (4)
LOSS OF CONTAINMENT INTEGRITY DUE TO A REACTOR CORE ISOLATION COOLING SYSTEM SINGLE FAILURE CRITERIA VIOLATION

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 NUMBERS(S)
0 3	1 8	9 3	9 3	-- 0 1 3	-- 0 0	0 4	1 9	9 3		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 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)	50.73(a)(2)(iv)	77.71(b)	
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.73(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)	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
C. D. Mackaman, Licensing Engineer	AREA CODE 5 0 9 3 7 7 - 4 4 5 1

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 (16)

On March 18, 1993, a design error was identified by Supply System engineers working in the Component Safety Classification Program. The Reactor Core Isolation Cooling (RCIC) system pump suction line from the suppression pool did not meet single failure design criteria. A failure of Division 1 DC power to the RCIC system would cause the motor operators for the Pump Suction Isolation Valve (RCIC-V-31) and the Auxiliary Cooling Water Supply Valve (RCIC-V-46) to fail in an as-is open position. This could result in an unisolable release of suppression pool water to the Reactor Building. This potential release path violates commitments made to General Design Criterion (GDC) 56 for isolation of Primary Containment. The failure to comply with GDC 56 resulted in Primary Containment being inoperable.

The root cause for this event was a deficiency in the original design of the Containment Isolation system.

Corrective actions for this event include: (1) The RCIC pump suction isolation valve from the suppression pool (RCIC-V-31) was de-energized and tagged closed to restore Primary Containment operability; (2) the RCIC system was declared inoperable; (3) emergency relief from Technical Specification 3.7.3 was requested and approved to allow continued operation; (4) perform an Engineering evaluation for a permanent resolution; (5) implement the Engineering evaluation recommendation(s); and (6) perform an Engineering review of containment penetrations that rely on single valve isolation.

Plant Conditions

Power Level - 100%
Plant Mode - 1 (Power)

Event Description

On March 18, 1993, a design error was identified by Supply System engineers working in the Component Safety Classification Program. The Design Requirement Document Program and the Component Safety Classification Program are ongoing design basis reconstitution programs funded through Fiscal Year 1995 and 1997, respectively. These programs provide a detailed review of WNP-2 design as a part of their methodology. The Component Safety Classification Program utilizes an Electric Power Research Institute (EPRI) Guideline based methodology to establish and document the technical basis for the safety classification of each safety-related component. The Design Requirement Document Program provides a documented reference to system level design requirements and functions.

A routine review of the safety-related containment portions of the Reactor Core Isolation Cooling (RCIC) system by Component Safety Classification engineers found that the containment isolation design for the RCIC pump (RCIC-P-1) suction line from the suppression pool did not meet single failure design criteria. A failure of Division 1 DC power to the RCIC system would cause the motor operators for the Pump Suction Isolation Valve (RCIC-V-31) and the Auxiliary Cooling Water Supply Valve (RCIC-V-46) to fail in an as-is open position. This single failure, combined with a Loss Of Coolant Accident (LOCA), results in an unisolable suppression pool water flow path from Primary Containment to the Reactor Building (see Figure 1). This potential release path violates commitments made to General Design Criterion (GDC) 56 for isolation of the RCIC-P-1 suction line containment penetration. The failure to comply with GDC 56 resulted in Primary Containment being inoperable.

Immediate Corrective Actions

Immediate Corrective actions for this event include:

1. On March 19, 1993, RCIC-V-31 was closed, and its motor operator was de-energized by tagging open the circuit breaker, to restore Primary Containment operability. This action removed the capability of RCIC to transfer RCIC-P-1 suction automatically to the suppression pool, although pump suction was still available from the Condensate Storage Tanks (CSTs). Since automatic transfer capability is a condition of operability as defined in Technical Specification 3.7.3, the RCIC system was declared inoperable at 1530 hours on March 19, 1993, and the associated Limiting Condition for Operation (LCO) was entered.

2. On March 26, 1993, the Supply System formally requested emergency relief from Technical Specification 3.7.3 to allow continued power operation until the 1993 Refueling Outage (R-8).

3. On April 1, 1993, the Supply System formally requested an emergency amendment to Technical Specification 3.7.3 to temporarily restore RCIC operability until the beginning of R-8.

4. At 1500 hours on April 2, 1993, WNP-2 received a Notice of Enforcement Discretion allowing continued power operation beyond the 14 day LCO of Technical Specification 3.7.3.

5. The RCIC system was declared operable at 2232 hours on April 9, 1993, based upon approval of the Technical Specification amendment request.

<u>Further Evaluation and Corrective Action</u>

A. <u>Further Evaluation</u>

1. In accordance with 10CFR50.72(b)(1)(ii)(B), this event was reported to the NRC Operations Center via the Emergency Notification System (ENS) at 1413 hours on March 19, 1993, as "Any event or condition during operation that . . . results in the nuclear power plant being . . . In a condition that is outside the design basis of the plant . . .. "

2. This event is reportable under 10CFR50.73(a)(2)(i)(B) as "Any operation or condition prohibited by Technical Specifications," and 10CFR50.73(a)(2)(ii)(B) as "Any event or condition . . . that resulted in the nuclear power plant being . . . In a condition that was outside the design basis of the plant . . .. "

3. A Loss Of Coolant Accident (LOCA) event scenario that could lead to the release of suppression pool water to the Reactor Building is as follows (see Figure 1):

   (1) The plant is at power with RCIC-P-1 suction in its normal alignment to the Condensate Storage Tank (CST).

   (2) A Small Break Loss Of Coolant Accident (SBLOCA) event occurs causing RCIC to initiate, or a Large Break Loss Of Coolant Accident (LBLOCA) event occurs causing RCIC-V-46 to open on low Reactor Pressure Vessel (RPV) water level.

   (3) During the event, RCIC-P-1 suction transfers from the CST to the suppression pool on low CST water level. This automatic suction transfer opens RCIC-V-31. For the LBLOCA event, the transfer could occur as a result of a seismically induced failure of the nonsafety-related CST.

(4) After the pump suction transfer, a loss of Division 1 DC power to the RCIC system occurs.

(5) The DC power failure causes a loss of RCIC control and power to the RCIC-V-31 and RCIC-V-46 motor operators resulting in both valves remaining open.

(6) The Lube Oil Cooler Pressure Control Valve (RCIC-PCV-15) will go full open attempting to maintain the downstream cooler pressure at approximately 50 psig.

(7) Suppression pool water will then drain through RCIC-V-31, RCIC-P-1, RCIC-V-46, RCIC-PCV-15, the lube oil cooler and the barometric condenser to the vacuum tank (see Figure 1). The elevation difference between the suppression pool water level and the vacuum tank will produce a static head sufficient to lift the vacuum tank relief valve at its setpoint of 6 psig.

(8) The suppression pool water will flow through the vacuum tank relief valve at approximately 5.5 gpm to the Equipment Drain (Radioactive) (EDR) system. The EDR system will then transfer the water to the Reactor Building equipment drain sump, where it will be collected. Since the sump is automatically isolated during a LOCA, it will fill and eventually overflow, flooding connecting rooms.

4. GDC 56 (10CFR50, Appendix A) requires each line that connects directly to the containment atmosphere and penetrates primary reactor containment to be provided with containment isolation valves. There must be one locked closed or automatic isolation valve inside containment and one locked closed or automatic isolation valve outside containment, or be demonstrated acceptable on some other defined basis. WNP-2 met containment isolation provisions for the RCIC-P-1 suction line containment penetration in accordance with prerequisites defined in Standard Review Plan (SRP) 6.2.4, Section II, Paragraph 6.e. Specifically, the provisions are; (1) System reliability must be greater with one isolation valve (instead of two) in the line; (2) The system must be closed outside containment and a single active failure can be accommodated with only one isolation valve; (3) This closed system must be protected from missiles; (4) The closed system must be designed to Seismic Category I, Safety Class 2 requirements, and a minimum temperature and pressure rating at least equal to that for the containment; and (5) The piping between the isolation valve and containment must be enclosed in the leak-tight housing or conservative design of the piping and valve, conforming to SRP 3.6.2, to preclude a breach of piping integrity.

The WNP-2 Final Safety Analysis Report (FSAR), Section 6.2.4.3.2.2.2.1, "HPCS, LPCS, RCIC, and RHR Suction Lines," provides justification for this exception to GDC 56 on the basis that RCIC system reliability is greater with only one isolation valve. Furthermore, the RCIC-P-1 suction piping is considered an extension of containment, and is designed to the same quality standards as containment. Although, SRP 6.2.4 and the FSAR provide acceptable basis for having only one isolation valve, a postulated single failure event causing RCIC-V-31 and RCIC-V-46 to simultaneously fail in an as-is open position was not addressed in the WNP-2 FSAR. Consequently, Primary Containment was outside the plant design basis, and technically inoperable during Operational Conditions 1, 2 and 3. Technical Specification 3.6.1.1 was violated when Primary Containment integrity was not maintained and became inoperable.

B. Root Cause

The root cause for this event was a deficiency in the original design of the Containment Isolation system. The single failure criterion for the postulated event scenario described above was not considered in the safety analysis.

A contributing cause was the failure to establish a corrective action for a similar event in February 1988, to look generically for similar concerns.

C. Further Corrective Action

1. No further corrective action associated with original contractor related design errors and deficiencies are being considered. Due to the detailed nature of the Design Requirement Document Program and Component Safety Classification Program engineering reviews, design errors and omissions of the type reported in this LER could be reasonably expected to be found. Because these programs are ongoing, there is the likelihood that additional reportable events will result. These events may be reported as revisions to this LER.

2. Perform an Engineering evaluation for a permanent resolution to ensure the RCIC-P-1 suction line meets single failure design criteria. This action to be completed by May 24, 1993.

3. Implement the RCIC-P-1 suction line single failure Engineering evaluation recommendation(s) before startup from the R-8 refueling outage, or obtain Technical Specification 3.7.3 relief to allow startup.

4. Perform an Engineering review of containment penetrations that rely on single valve isolation to determine if single failure related unisolable flow paths from containment exist. This action to be completed by September 30, 1993.

Safety Significance

The FSAR, Section 6.3.3, Emergency Core Cooling System (ECCS) Performance Evaluation, shows that a SBLOCA event does not lead to fuel damage. Consequently, the postulated DC power failure event scenario described above would only release reactor coolant activation and corrosion products to the suppression pool water and Reactor Building equipment drain sump. This limited release of radionuclides would be treated by the Standby Gas Treatment (SGT) system before release to the environment.

According to FSAR, Section 15.6.5.5.1.1, Fission Product Release from Fuel, a postulated LBLOCA and Division 1 DC power failure event scenario would result in significant fission product release from the fuel. According to FSAR, Section 15.6.5.5.1.2, Fission Product Transport to the Environment, some of the release would be to the suppression pool. While the potential release to Secondary Containment would be treated by the SGT system, the fission product transport to the environment could have safety significance.

A Division 1 DC power failure event scenario involving fuel damage would only result in a limited release of radionuclides to Secondary Containment. The postulated flow path to the Reactor Building equipment drain sump would be isolated by Annunciator Response Procedure PPM 4.601.A4, 601.A4 Annunciator Panel Alarms, and Emergency Operating Procedure (EOP) PPM 5.3.1, Secondary Containment Control. This would result in action to close RCIC-V-31 (by remote manual operation) when the RCIC Pump Room water level reached six inches (the maximum safe operating value). The airborne release would also be treated by the SGT system before release to the environment.

Similar Events

On February 19, 1988, LER 88-002 reported an original plant design error that caused the RCIC-P-1 suction line not to meet containment isolation and single failure criteria. In February 1988, the Supply System had not implemented a formal Root Cause Analysis Program, as it was still in the formative stages. Consequently, no corrective action was identified in the LER or the associated Nonconformance Report (288-0022) that would have prevented a similar occurrence.

The Supply System implemented a Root Cause Analysis Program in December 1988, in accordance with Plant Administrative Procedure PPM 1.3.48, Root Cause Analysis, that addresses similar generic concerns to prevent recurrence.

EIIS Information

Text Reference	EIIS Reference System	EIIS Reference Component
Reactor Core Isolation System (RCIC)	BN	---
RCIC System Suction Line	BN	PSP
Condensate Storage Tank (CST)	KA	TK
RCIC Pump 1 (RCIC-P-1)	BN	P
Suppression Pool Suction Valve (RCIC-V-31)	BN	ISV
Auxiliary Cooling Water Supply Valve (RCIC-V-46)	BN	ISV
RCIC Motor Operated Valves	BN	20
Primary Containment	NH	---
Secondary Containment	NH	---
Reactor Building	NG	---
Pressure Control Valve (RCIC-PCV-15)	BN	PCV
RCIC Lube Oil Cooler	BN	CLR
RCIC Barometric Condenser	BN	CDU
Equipment Drain (Radioactive) (EDR) System	WK	---
Equipment Drain Sump	WK	TK
DC Power	EJ	JX
Standby Gas Treatment (SGT) System	BH	---

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TITLE (4)
LOSS OF CONTAINMENT INTEGRITY DUE TO A REACTOR CORE ISOLATION COOLING SYSTEM SINGLE FAILURE CRITERIA VIOLATION

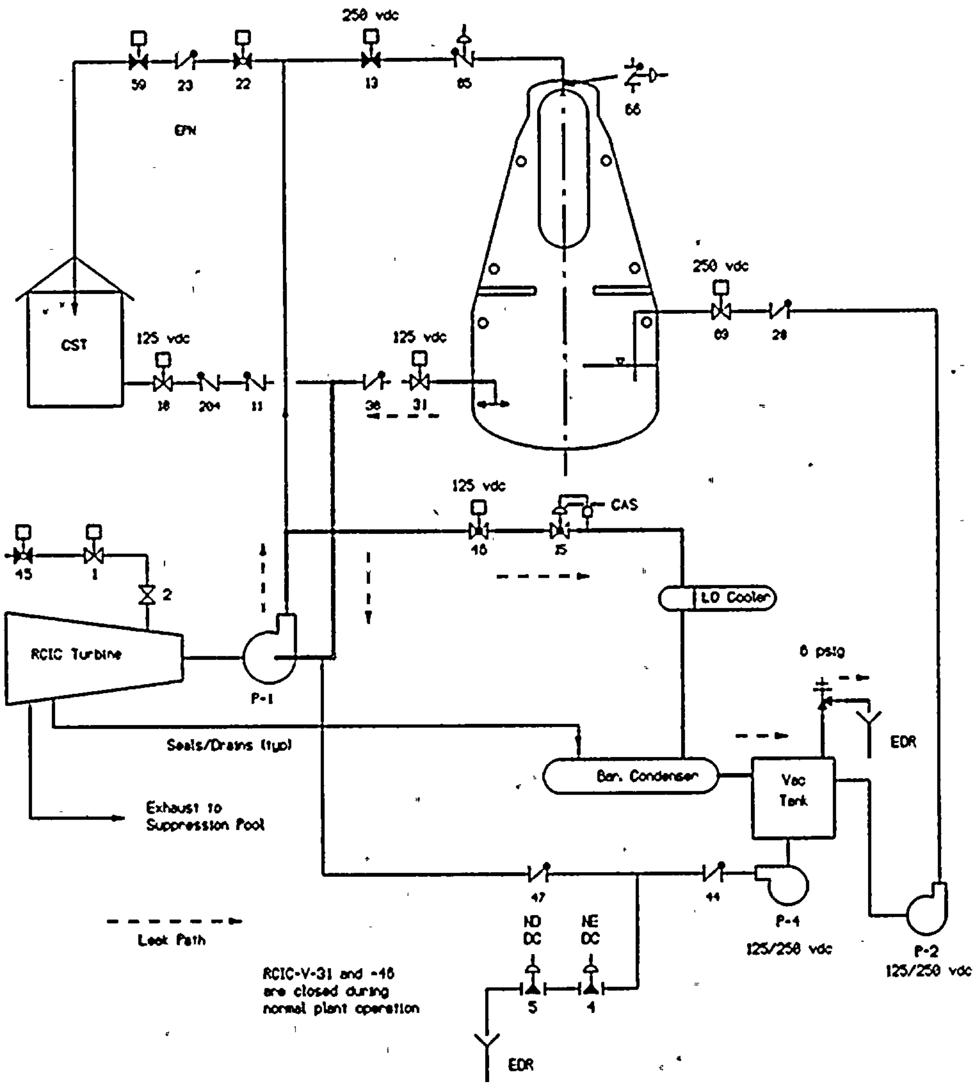

Figure 1