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SUBJECT: LER 94-020-00: on 941111, postulated that MCR fire could induce hot short that could result in spurious closure of MO CI valve CIA-V-20. Caused by inadequate analysis. Continuous fire watch established for area. W/941212 ltr.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

December 12, 1994
GO2-94-277

Docket No. 50-397

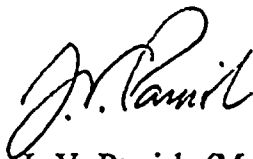
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Subject: **NUCLEAR PLANT WNP-2, OPERATING LICENSE NPF-21
LICENSEE EVENT REPORT NO. 94-020**

Transmitted herewith is Licensee Event Report No. 94-020 for the WNP-2 Plant. This report is submitted in response to the reporting requirements of 10CFR50.73 and discusses the items of reportability, corrective action taken, and action taken to preclude recurrence.

Should you have any questions or desire additional information, please call me or D.A. Swank at (509) 377-4278.

Sincerely,



J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

JVP/CJF/jcs
Enclosure

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LICENSEE EVENT REPORT (LER)

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Washington Nuclear Plant - Unit 2

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(4) POTENTIAL OF SPURIOUS ACTUATION OF PNEUMATIC SUPPLY VALVE TO IMPAIR OPERATION OF MAIN STEAM SAFETY RELIEF VALVES

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)	
1	1	1	9	4	0	2	0	0	0	0	
1	1	1	9	4	0	2	0	0	0	0	

OPERATING MODE (9) X THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

POWER LEVEL (10)	20.402(b)	20.405(c)	50.73(a)(2)(iv)	77.71(b)
1	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.73(c)
0	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)
0	20.405(a)(1)(iii)	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)	X 50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
C.J. Foley, Licensing Engineer	AREA CODE 5 0 9 3 7 7 - 4 3 2 5

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
NA									
NA									

SUPPLEMENTAL REPORT EXPECTED (14)

EXPECTED SUBMISSION DATE (15)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) X ☐ NO

ABSTRACT

On November 11, 1994, WNP-2 was operating at 100% power in Mode 1. An engineer postulated that a Main Control Room fire could induce a hot short that could result in spurious closure of motor-operated containment isolation valve CIA-V-20. This might result in depletion of pneumatic supplies for Main Steam Relief Valves during the interval between evacuation of the Control Room and assumption of plant control from remote shutdown panels. No fire actually occurred. The NRC was notified of this event at 1726 (PST) pursuant to 10CFR50.72(b)(2)(iii)(A). Subsequent analysis revealed that the event was due to inadequate analysis rather than a condition that would have prevented maintaining the reactor safely shut down in event of a Control Room fire and should be reported only under 10CFR50.73(a)(2)(iii)(A). The cause of the event was a design deficiency due to lack of adequate consideration of the plant response during the interval between Control Room evacuation due to a fire and assumption of plant control from the remote shutdown panels.

A continuous fire watch was established for the area, and a procedure revised to assure an operator would be immediately prestaged at a remote shutdown panel to transfer control of certain Main Steam Relief Valves from the Control Room to a remote shutdown panel in event of a Control Room fire. A design change will be implemented to eliminate the problem, and a procedure governing analysis of equipment used for safe shutdown after a Control Room fire will be changed to require consideration of on-going plant processes during the interval between Control Room evacuation and assumption of control from remote panels.

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Event Description

On November 11, 1994, WNP-2 was operating at 100% power in Mode 1. As a result of a review of the design and layout of controls and conductors located in the Control Room, an engineer postulated that a fire in the Main Control Room could induce a hot short in a control switch [HS] or an electrical cable [CBL], that could result in spurious closure of motor-operated containment isolation valve [ISV] CIA-V-20. This valve provides compressed nitrogen to eighteen accumulators [ACC] inside of primary containment that are associated with the eighteen Main Steam Safety Relief Valves (MSRVs) [SB] [RV]. Spurious closure of this valve could result in the inability to manually activate 3 of 6 MSRVs from a remote location, as could be necessary if the Main Control Room had to be evacuated due to the postulated fire.

The NRC was notified of this event at 1726 (PST) pursuant to 10CFR50.72(b)(2)(iii)(A).

Further Evaluation

In the event that the Main Control Room must be evacuated because of a Control Room fire, the Reactor must be depressurized from remote shutdown panels to permit cooling to be established by Low Pressure Core Injection (LPCI) [BO]. Three MSRVs must be opened from the Remote Shutdown (RSD) panel, and three from the Alternate Remote Shutdown (ARSD) panel, using switches on these panels. MSRV design is such that compressed gas is used to overcome actuator spring forces when the MSRV is opened in either relief or manual mode. Each MSRV is equipped with pneumatic accumulators [TK] to provide the compressed gas. The accumulators are serviced by a compressed nitrogen supply located external to the primary containment. The three MSRVs controlled from the Remote Shutdown (RSD) panel are designed to be part of the Automatic Depressurization System (ADS), the three that may be controlled from the ARSD panel are nonADS. ADS MSRVs are equipped with two accumulators, a larger one for use only in ADS mode, and a smaller one for use in ordinary relief mode. NonADS MSRVs are equipped with only the smaller sized accumulators for use in ordinary relief mode. The potential spurious closure of motor-operated containment isolation valve CIA-V-20 would isolate the smaller accumulators for all eighteen MSRVs from the external compressed nitrogen supply. During Control Room evacuation, the MSRVs may open cyclically in the relief mode, resulting in rapid depletion of the isolated accumulator for the particular MSRV. The MSRV(s) would then close until Reactor pressure rose to the safety setting, at which level the Reactor pressure is sufficient to overcome MSRV spring actuator forces. The MSRVs would then reopen, maintaining Reactor pressure at the safety setting level. While Reactor overpressure is thereby prevented, lack of compressed nitrogen in the smaller accumulators would prevent manually opening the three nonADS MSRVs from the ARSD panel, delaying initiation of LPCI. The other three MSRVs are ADS equipped. The larger ADS accumulators would not be depleted by MSRV cycling in relief mode. Consequently, the three ADS MSRVs could still be opened from the RSD panel to depressurize the Reactor to enable cooling by LPCI. However, the current commitment requires six MSRVs to be opened to assure depressurization in time to prevent peak fuel cladding temperatures from exceeding approved limits. Consequently, safe shutdown of the Reactor in strict adherence to the currently authorized analysis would be jeopardized if a

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fire induced hot short caused spurious closure of the CIA motor-operated containment isolation valve providing compressed nitrogen service to accumulators for the three MSRVs to be opened from the ARSD panel.

The basic design of the RSD/ARSD panels was established during original construction, at which time a choice was made to control 3 nonADS MSRVs from the ARSD panel. No bases has been found for that choice. However, if ADS MSRVs had been chosen for both the RSD and ARSD panels, the situation outlined above would not be possible, because six ADS MSRVs would be available on the remote panels. This constitutes a design error based on lack of adequate consideration of plant response during the interval between Control Room evacuation and assumption of control from remote shutdown panels. A safe shutdown analysis was performed by the original architect engineer, and was updated by the Supply System in 1990. Consideration was given to the capacity of the accumulators in an analysis in 1988, and again in 1994. However, none of these analyses accurately accounted for the plant response during the time interval between Control Room evacuation and assumption of plant control from the RSD/ARSD panels. This time interval is between 5 and 10 minutes, while accumulator depletion could occur in approximately 7.2 minutes if MSRV cycling began immediately upon initiation of a Control Room fire that spuriously closed CIA-V-20. Therefore, the cause of the event was a design deficiency during the original construction phase, resulting from analysis that did not adequately consider the plant response during the time interval intervening between Control Room evacuation due to a fire and assumption of plant control from the remote shutdown panels.

Corrective Action

A continuous fire watch was immediately initiated to assure early detection of a Main Control Room fire. The procedure governing safe shutdown for a Control Room fire scenario was changed to require immediate prestaging of an operator to the ARSD panel to transfer control of the three nonADS MSRVs from the Main Control Room to the ARSD panel. This would have the effect of putting those MSRVs on manual control from the ARSD panel, preventing automatically initiated use of the compressed nitrogen in those three accumulators.

To address the lack of consideration of the time interval between Control Room evacuation and assumption of control of the plant from the RSD/ARSD panels, the procedure defining the process to be followed for safe shutdown analyses will be augmented to require explicit consideration of time needed for operators to react to fire conditions. The current design review will be completed to assure no similar problems are present.

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A plant design change will be implemented during the refueling outage scheduled for the Spring of 1995 to alter the controls on the ARSD panel such that the existing switches will control three ADS MSRVs instead of three nonADS MSRVs. After implementation, it will be possible to open a total of six ADS MSRVs from outside the Control Room. This will assure that adequate compressed nitrogen is available to open these MSRVs from the RSD/ARSD panels under postulated circumstances.

Safety Significance

Analysis performed by General Electric (GE) in support of WNP-2 power uprate indicates that reactor depressurization may be effected with 2 MSRVs while maintaining peak fuel clad temperatures (PCT) slightly below 1500°F, the threshold for fuel damage. In this specific case, the core power was not at power uprate conditions, but at approximately 95% of that level. Consequently, the post-SCRAM thermal levels were also at 95% of the level assumed for the GE analysis; this would have the effect of reducing PCT below the predictions of the GE analysis. Three ADS MSRVs were actually available to depressurize the reactor for the Control Room fire scenario. The peak fuel cladding temperature would be lower for depressurization for 3 MSRVs than for 2 MSRVs. As a point of reference, the GE analysis also considers a 6 MSRV case. The predicted PCT for that case is only 588°F. While no analysis is available for a 3 MSRV case, the two cases available bound the actual situation. It may therefore be concluded that PCT would not have closely approached the 1500°F threshold for fuel damage if a recovery from an actual Control Room fire had to be effected.

Additionally, the probability of occurrence of Control Room fires that would result in both a Control Room evacuation and concurrent fire damage to the control circuits for valve CIA-V-20 is very low. The event arose from an examination of the possibilities of fire damage to equipment in the Control Room. No actual fire occurred.

Consequently, this event was not safety significant because if a fire had occurred in spite of the very low probability of occurrence of such an event, equipment was available to effect safe shutdown of the plant. No radioactivity would have been released because the peak fuel cladding temperatures would have been maintained below the threshold of actual fuel damage.

Similar Events

LER 94-010 documented inadequate analysis of the effect of a control room fire on circuitry supporting operation of the Residual Heat Removal System when initiated as necessary to safely shut down the plant after a fire.

