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April 29, 1996

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 96-07-00

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

The attached Licensee Event Report LER 96-07-00 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Very truly yours,

Stephen E. Quinn

Attachment

cc: Mr. Thomas T. Martin
Regional Administrator - Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Jefferey F. Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
Buchanan, NY 10511

NRC FORM 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH 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.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Indian Point Unit No. 2	0 5 0 0 0 2 4 7	1 OF 0 4

TITLE (4)

Potential Challenge of High/Low Pressure Interface [Appendix R]

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 5 0 0 0
0 3	2 9	9 6	9 6	– 0 0 7	–	0 0	0 4	2 9	9 6		0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
N		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
1 0 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)
		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
George Dahl, Engineer	AREA CODE 9 1 4 7 3 4 – 5 1 8 6

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)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
XX YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 8	0 5	9 6

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On March 29, 1996, it was determined based upon the preliminary results of an engineering re-analysis, that the facility may be in a condition that is outside its licensing design basis for the facility fire protection program (10 CFR 50, Appendix R). For assumed fires in certain areas, the potential for hot shorts may exist which may challenge a reactor coolant system high/low pressure interface (Pressurizer PORV/block valve). Until the analysis can be completed and reviewed, compensatory actions of a fire watch and procedure change have been put in place.

PLANT AND SYSTEM IDENTIFICATION:

Westinghouse 4-Loop Pressurized Water Reactor

IDENTIFICATION OF OCCURRENCE:

For assumed fires in certain areas, the potential for hot shorts may exist which may challenge a reactor coolant system high/low pressure interface (Pressurizer PORV/block valve).

EVENT DATE:

March 29, 1996

REPORT DUE DATE:

April 29, 1996

REFERENCES:

Significant Occurrence Report (SOR) 96-309

PAST SIMILAR OCCURRENCES:

None

DESCRIPTION OF OCCURRENCE:

On March 29, 1996, with the plant at 100% power, a consultant presented the preliminary results of an engineering re-analysis of the implications to the fire protection program of different operating positions of the pressurizer block valves and power operated relief valves (PORVs). The issue involves potential opening of a PORV in the event of certain fire scenarios due to postulated hot shorts of its DC control circuit from normally energized circuits. The same fire is also postulated to prevent the closure of the PORV's associated block valve due to three phase hot shorting of its AC power circuit or AC control circuit from normally energized circuits, or due to the opening of its power or control circuit because of fire damage. It was decided to implement several compensatory actions prior to the finalizing of the re-analysis.

ANALYSIS OF OCCURRENCE:

The pressurizer at Indian Point Unit No. 2 is provided with two power operated relief valves (PORVs) for automatic and manual reactor coolant system pressure control. Each PORV line is provided with a motor-operated block valve, and the two lines relieve to the pressurizer relief tank (PRT). The PORVs are controlled by 125VDC solenoid operated valves (SOVs) that provide nitrogen to open the valves. The block valve motor operators are powered by 480VAC and must remain energized to move in the open or close position, and control and indication for the valves are powered by 120VAC. The PORV/block valve paths are high/low pressure interfaces with the reactor coolant system (RCS), with approximately 2235 psig on the pressurizer side and slightly greater than atmospheric pressure on the PRT side. The PORV is normally closed and the block valve is normally open during power operation.

Although not specifically addressed in Appendix R, NRC guidance documents specify that where adequate separation of cabling for the redundant valves in the line comprising the reactor coolant system high/low pressure interface is not provided, it must be demonstrated that fire induced failures (hot short, open circuits, or short to ground) of the cables will not cause maloperation of the valves.

The cables for both PORV SOVs and their associated block valves are routed in the same fire area and are contained within trays that contain cables of similar size and voltage which are normally energized. In the event of a fire, it is theoretically postulated that the insulation of a PORV SOV cable burns in a manner to expose its two control circuit conductors, and the cable of a normally energized circuit also burns in a manner to expose its two conductors. The four conductors then physically contact such that the normally energized conductors provide power to the PORV SOV control circuit. A PORV is thus opened and the operators cannot close it because the normally energized circuit has locked it open. Since the block valve is already open, reactor coolant is able to flow out of the pressurizer, into the PRT, and into containment once the rupture disk opens. Closure of the block valve would stop the loss of reactor coolant through this path. However, the block valve power and control cables are routed in trays which contain normally energized cables of similar size and voltage. Therefore, the same fire is postulated to hot short the block valve power or control circuit with the normally energized circuit of another component, or to damage either the power or control circuit such that a close signal is not able to reach the valve motor operator.

As this potential fire-initiated scenario may not satisfy NRC guidance regarding reactor coolant system high/low pressure interfaces, this represents a condition that may be outside the licensing basis of the plant and is reportable per 10 CFR 50.72(b)(1)(ii)(B) and 10 CFR 50.73(a)(2)(ii)(B). Previously, conservative bounding fire effects thermal analyses were performed to demonstrate that the PORV SOV cables, the block valve cables, and normally energized cables in the same

ANALYSIS OF OCCURRENCE (continued):

cable trays would be adequately protected by their jackets such that they will not fail in a manner to expose conductors that can cause phase-to-phase hot shorts. Therefore, a fire-initiated loss of coolant would not be possible and it would not matter if the block valves are open or closed during power operation. It is believed that these analyses provide assurance that this RCS high/low pressure interface would not have been compromised in the event of a fire so that the safety significance of this condition is minimal.

CAUSE OF OCCURRENCE:

Initially, it was believed that a change in the normal operating position of the block valves may be contrary to the assumptions in the fire protection program plan. At this time, it is not clear if the evaluation which performed the conservative bounding fire effects thermal analyses referred to above conforms to the guidance provided.

CORRECTIVE ACTIONS:

Upon review of the preliminary results of this recent re-analysis, compensatory actions were taken prior to the finalizing of the analysis. One-hour fire watch tours were instituted in plant areas outside containment traversed by power and control cables for the PORVs and the block valves, increased (once per hour) monitoring of containment air temperature was instituted, and a temporary procedure change was implemented to require closing of the block valves prior to exit from the control room in the event a fire prevents operation of safe shutdown equipment from the control room or the control room is rendered uninhabitable.

The final consultant analysis is currently undergoing an independent internal review. Compensatory actions will remain in place until this review is complete. Due to the large number of cables (930) whose routing needs to be reevaluated, this review is expected to be completed in June 1996. Subsequent to that, a supplemental report will be provided identifying the results, the need to continue the compensatory actions, or any other actions to be taken.