NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO: Mr. J. G. Keppler	FROM: Indiana & Michigan Power Corp. Brtdgman, Mich. R. W. Jurgensen	DOCKET NUMBER 50-315
		FILE NUMBER INCIDENT REPORT
		DATE OF DOCUMENT 7/1/76
		DATE RECEIVED 7/6/76
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DESCRIPTION

Ltr. trans the following:

(1-P)

PLANT NAME:

Cook #1

ENCLOSURE

Licensee Event Report (RO 50-315/76-28) on 6/17/76 concerning the ranges of the pressurizer pressure transmitters being changed.

ACKNOWLEDGED

DO NOT REMOVE

(4-P)

NOTE: IF PERSONNEL EXPOSURE IS INVOLVED SEND DIRECTLY TO KREGER/J. COLLINS

SAFETY — FOR ACTION/INFORMATION — ENVIRO 7/6/76 RJL

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NRC FORM 195 (2-76)

INDIANA & MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT
P.O. Box 458, Bridgman, Michigan 49106

RECEIVED
JUL 6 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

July 1, 1976

Mr. J.G. Keppler, Regional Director
Office of Inspection and Enforcement
United States Nuclear Regulatory Commission
Region III
799 Roosevelt Road
Glen Ellyn, IL 60137

Operating License DPR-58
Docket No. 50-315

Dear Mr. Keppler:

Pursuant to the requirements of Appendix A Technical Specifications and the United States Nuclear Regulatory Commission Regulatory Guide 1.16, Revision 4, Section 2.a, the following report is submitted:

RO 50-315/76-28

Sincerely,

R.W. Jurgensen
Plant Manager

Regulatory Docket File

/bab

DOCKETED USNRC
JUL 6 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

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LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (PLEASE PRINT ALL REQUIRED INFORMATION)

	LICENSEE NAME	LICENSE NUMBER	LICENSE TYPE	EVENT TYPE
01	M I D C C 1	0 0 - 0 0 0 0 0 - 0 0	4 1 1 1 1	0 1

	CATEGORY	REPORT TYPE	REPORT SOURCE	DOCKET NUMBER	EVENT DATE	REPORT DATE
01 CON'T		T	I	0 5 0 - 0 3 1 5	0 6 1 7 7 6	0 7 0 1 7 6

EVENT DESCRIPTION

02 WHILE IN MODE 1 AT 100 PERCENT POWER, DURING AN AUDIT, THE NRC RESIDENT INSPECTOR FOUND

03 THAT IN APRIL OF 1976 THE RANGES OF THE PRESSURIZER PRESSURE TRANSMITTERS HAD BEEN CHANGED

04 FROM 1674.1 - 2471.1 PSI TO 1725.9 - 2525.9 PSI. THIS WAS DONE TO RECTIFY AN IMPROPER

05 HEAD CORRECTION.

06 (SEE SUPPLEMENT) (RO-050-0315-76-28)

	SYSTEM CODE	CAUSE CODE	COMPONENT CODE	PRIME COMPONENT SUPPLIER	COMPONENT MANUFACTURER	VIOLATION
07	I A	D	I N S T R U	N	W 1 2 0	Y

CAUSE DESCRIPTION

08 SEE SUPPLEMENT

09

10

	FACILITY STATUS	% POWER	OTHER STATUS	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION
11	E	1 0 0	NA	D	NRC INSPECTORS AUDIT

	FORM OF ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY	LOCATION OF RELEASE
12	Z	Z	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION
13	0 0 0	Z	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION
14	0 0 0	NA

PROBABLE CONSEQUENCES

15 NA

LOSS OR DAMAGE TO FACILITY

	TYPE	DESCRIPTION
16	Z	NA

PUBLICITY

17 NA

ADDITIONAL FACTORS

18 SEE SUPPLEMENT

19

NAME: J. RISCHLING PHONE: (616)465-5901 (368)

SUPPLEMENT TO REPORTABLE OCCURANCE RO-050-0315-76-

On April 29, 1976, during a scheduled plant outage, it was discovered by a Control and Instrument Senior Technician that a 25 psi static head correction to the pressurizer pressure transmitters (NPP-151, NPP-152, NPP-153, NPS-153) had been subtracted from the transmitter input instead of being added. This error had the effect of lowering the setpoint for the pressurizer low pressure reactor trip from 1865 psig to 1815 psig (Technical Specification Section 2.2.1) and the pressurizer low pressure portion of the safety injection and feedwater isolation signal from 1815 psig to 1765 psig (Technical Specifications Section 3.3.2.1). The error also resulted in the operating band for pressurizer pressure being maintained at a pressure 50 psi lower than the normal operating pressure as specified in Technical Specifications Section 3.2.5. As soon as the error was discovered, all four pressurizer pressure transmitters were recalibrated correctly and temporary change sheets to the calibration procedures were prepared. Also, a complete review of all calibration procedures containing pressure head corrections was made with no other errors found.

On January 19, 1974, the initial calibration of the pressurizer pressure transmitters was performed. Prior to this date, discussions were held to determine if the 25 psi static head correction should be added or subtracted. As a result of these discussions, it was erroneously decided that this static head should be subtracted from the transmitter span to allow for proper head correction. The calibrations were repeated in October 1974, and again in April 1976, at which time the error was found. Although the error was promptly corrected, a condition report was inadvertently not prepared which would have initiated NRC notification.

A review has been made of the effects on safety of the plant with incorrect pressurizer pressure transmitter setpoints. Operation in this manner has three effects with respect to the safety analysis, which are, as mentioned above (1) an effective lowering of the trip points for low pressurizer pressure reactor trip from 1865 to 1815 psig, (2) lowering the trip point for safety injection actuation and feedwater isolation from 1815 psig to 1765 psig, and (3) actual operation of the plant could have been as low as 50 psi below that pressure specified in the technical specifications which is the initial pressure assumed in the safety analysis.

Operation of the unit with the lower reactor coolant pressure over the time period involved did not cause any conditions which had an adverse affect on the health and safety of the public. The effects on the safety analyses of the lower pressurizer pressure trip setpoint and lower reactor coolant pressure are discussed below.

In Section 14 of the Donald C. Cook Nuclear Plant FSAR, the safety analyses assume reactor trip to occur at 1685 psig rather than the low pressure setpoint of 1865 psig. The reason for this difference is to allow for setpoint error. In practice, the actual gages were calibrated with weights traceable to the National Bureau of Standards, and the maximum difference between actual pressure and indicated gage pressure is conservatively estimated to be no greater than 10 psi. Thus, a low pressurizer pressure reactor trip setpoint of 1865 psi would reflect an actual pressurizer pressure reactor trip at no less than 1805 psi, assuming calibration errors resulted in the lowest possible system pressures. Thus, during operation with this incorrect setpoint, the low pressure setpoint would still have resulted in reactor trip at a level above the minimum analyzed in the safety analysis.

Operation with this incorrect setpoint resulted in the plant being slightly outside the initial pressure bounds assumed in both the loss of coolant accident analysis (LOCA) and plant transient analysis. With respect to LOCA analysis, operating at an initial system pressure 50 psi lower than the pressure assumed in the analysis has a minor effect in the subcooled blowdown portion of the analysis. This effect is in the conservative direction in that ECCS analyses are based on high initial system pressures in order to obtain conservatively high peak clad temperatures. The short term containment peak pressures would have also been lowered by some very small amount since subcooled blowdown decreases with decreasing system pressure, thus lowering the energy deposition into the containment. Long term pressure would not be affected by this small change in system pressure.

Lowering of system pressure and safety system setpoints by 50 psi had the effect of lowering the departure from nucleate boiling ratio (DNBR) associated with various plant transients analyzed in Section 14.1 of the FSAR. An assessment of the effect of lowering system pressure by 50 psi was made by evaluating the effect on the correlation used to evaluate DNBR (the W-3 correlation). As a result of this evaluation, in conjunction with a review of DNB data, it is concluded that the worst effect on system transients would have been to lower the calculated DNBR by approximately 3 percent.

A review of the limiting DNBR for the various transients and accidents included in the safety analysis in both Section 14 of the Donald C. Cook Nuclear Plant FSAR and the Fuel Densification Report (Reference 1) was made. This review indicated that these analyses included the effect of a densification power spike, and was completed prior to submission of Reference 2, which indicates that the densification power spike is not appropriate to include as part of DNB calculation. If the accident analysis in the FSAR and Reference 1 had been redone on the basis of information presented in Reference 2, the result would have been to raise the calculated DNBR by approximately 5 percent, thus completely offsetting the effect of operating the plant at 50 psi below the pressure specified in the Technical Specifications.

As a result of the above, it is concluded that if any of the postulated accidents analyzed in the FSAR should have occurred with this error in effect the health and safety of the public would not have been adversely affected.

To prevent reoccurrence of this error, the instrument maintenance procedures governing the calibration of these transmitters have been revised. Also, control and instrument personnel were reminded to insure that errors of this type are promptly reported, and a program has been initiated to ensure instrument maintenance procedures include a signoff on the data sheets indicating that any out-of-specification data has been reported.

In April, 1976, the temporary sheet to the pressurizer pressure transmitter calibration procedure was reviewed and approved by the Plant Nuclear Safety Review Committee (PNSRC). However, the PNSRC overlooked the fact that the setpoint error had been a technical specification violation. To give more assurance that temporary sheets are reviewed for their impact on the technical specifications, a check list will be used by the PNSRC which will include an item requiring evaluating the effect of the change on technical specifications.

A check list has been prepared by the plant and is under PNSRC review. Use of a final check list by the PNSRC will be initiated by July 30, 1976.

REFERENCES:

1. Westinghouse Report WCAP-8164, "Fuel Densification -- Donald C. Cook Nuclear Plant Units 1 and 2", dated June, 1973.

2. Westinghouse Report WCAP-8218, "Fuel Densification -- Experimental Results and Model for Reactor Application", dated October, 1973.