

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 7904040146 DOC. DATE: 79/03/21 NOTARIZED: NO DOCKET #
 FACIL: 50-316 DONALD C. COOK NUCLEAR POWER PLANT, UNIT 2, INDIANA & 05000316
 AUTH. NAME AUTHOR AFFILIATION
 SHALLER, D.V. INDIANA & MICHIGAN POWER CO.
 RECIP. NAME RECIPIENT AFFILIATION
 KEPPLER, J.G. REGION 3, CHICAGO, OFFICE OF THE DIRECTOR

SUBJECT: FORWARDS 790106 SAFETY INJECTION ACTUATION (SPECIAL REPT
 SI-03). STEAM LINE PRESSURES INDICATED HIGH, CAUSED BY FROZEN
 STEAM PRESSURE INDICATION LINES FORCING INDICATED PRESSURE
 100 PSI GREATER THAN OTHER STEAM PRESSURE INDICATIONS.

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	14 TA/EDO	1	1	15 NOVAK/KNIEL	1	1
	16 EEB	1	1	17 AD FOR ENGR	1	1
	18 PLANT SYS BR	1	1	19 I&C SYS BR	1	1
	20 AD PLANT SYS	1	1	21 AD SYS/PROJ	1	1
	22 REAC SAFT BR	1	1	23 ENGR BR	1	1
	24 KREGER	1	1	25 PWR SYS BR	1	1
	26 AD/SITE ANAL	1	1	27 OPERA LIC BR	1	1
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	29 ACRS	16	16			

APR 5 1980
 REG. J. JAMES

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INDIANA & MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT
P.O. Box 458, Bridgman, Michigan 49106

March 21, 1979

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-74
Docket No. 50-316
Special Report: SI-03

Dear Mr. Keppler:

The purpose of this letter is to forward to you the attached Special Reports required by Appendix A Technical Specification 3.5.2, Emergency Core Cooling Subsystems.

Sincerely,

D. V. Shaller
Plant Manager

/mkm

cc: J. E. Dolan
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INDIANA AND MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT

Operating License DPR-74
Docket No. 50-316
Special Report: SI-03

SAFETY INJECTION ACTUATION - JANUARY 6, 1979

CONDITIONS PRIOR TO OCCURRENCE

The Reactor was in operational Mode 1 at 100% steady state power. Control Rods were in the manual mode of operation. Steam Generator level controls and Pressurizer Pressure and level controls were in the automatic mode of control and functioning properly. All Control Rods were fully withdrawn except Control Bank "D" which was at 217 steps. Reactor Coolant System Boron Concentration was 674 PPM.

DESCRIPTION OF OCCURRENCE

Steam Line Pressures from #2 and #3 Steam Generators started indicating high. While the Steam Lines, Steam Generator Stop Valves, and Pressure Sensing Lines are in an enclosure, the enclosure is designed with vents and blow-out panels in case of Steam Line rupture. Outside air temperature was -60°F. With these two pressure indications increasing and all other parameters constant, the analysis was that these two sensing lines were freezing up and steps were being taken to thaw them out.

Before thaw-out could be accomplished, the two frozen lines increased pressure to the Transmitters and Reactor Protection by 100 psi greater than a normal operating Steam Line Pressure.

DESIGNATION OF CAUSE OF OCCURRENCE

The two frozen Steam Pressure Indication Lines forced the indicated pressure 100 psi greater than the other steam pressure indications. This allowed the 100 psi high Steam Line Differential on 2/3 Pressure Channels on 1/4 Steam Generators Logic to complete and safety injection was actuated as designed.

ANALYSIS OF OCCURRENCE

The following is a list of major items that were reviewed for their safety injections:

(a) Reactor Coolant System Cooldown Rate

The cooldown rate was an almost instantaneous drop on Tavg from 573°F to 540°F due to the Reactor-Turbine trip and actuation of the Steam Dump System. The rapid cooldown was terminated when the Steam Dump Valves were automatically tripped closed when Lo-Lo Tavg of 541°F was reached. The overall cooldown rate was similar to that experienced during a Reactor trip without safety injection.

(b) Thermal Effects of Safety Injection

During this occurrence the Centrifugal Charging Pumps injected to the Reactor through the safety injection path for approximately 14 minutes. During this time approximately 4,200 gallons of borated water with an initial temperature of 165°F was injected. This is the third inadvertent safety injection in which water was injected into the Reactor Coolant System and conservatively would constitute less than 3.2/10,000 of allowable cycles. The total accumulated cycles to date are 6.5/10,000.

(c) Effects on the Emergency Core Cooling System Piping (ECCS)

The piping and supports in the ECCS were given a thorough visual inspection to determine if any mechanical damage was experienced during the safety injection. There was no evidence of any mechanical damage or abnormal movements of the piping.

CORRECTIVE ACTIONS

The frozen Pressure Sensing Lines were thawed. Some vents in the enclosure that were open to the outside were sealed over with plastic film. The Plant has a winterizing checklist and these vents have been added to this list to prevent reoccurrence.