CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9705270282 DOC.DATE: 97/05/20 NOTARIZED: YES DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana M 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana M 05000316

AUTH.NAME AUTHOR AFFILIATION
FITZPATRICK,E. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

SUBJECT: Informs that analyses have shown thermally induced pressurization of listed lines during design basis accident will not result in piping stresses that exceed FSAR allowables,per Generic Ltr 96-06.

DISTRIBUTION CODE: A072D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 5
TITLE: GL 96-06, "Assurance of Equip Oprblty & Contain.Integ. during Design

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	NRR/WETZEL,B.	1	1	PD3-3 PD	1	1
	HICKMAN,J	1	1			
INTERNAL:	~~FILE CENTER 01~~	1	1	NRR/DE/EMEB	1	1
	NRR/DSSA/SCSB	1	1	NRR/DSSA/SPLB	1	1
EXTERNAL:	NOAC	1	1	NRC PDR	1	1

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 9 ENCL 9

Indiana Michigan
Power Company
500 Circle Drive
Buchanan, MI 49107 1395

AEP
INDIANA
MICHIGAN
POWER

May 20, 1997

AEP:NRC:1256B

Docket Nos.: 50-315
50-316

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

Gentlemen:

Donald C. Cook Nuclear Plant Units 1 and 2
NRC GENERIC LETTER 96-06
ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT
INTEGRITY DURING DESIGN BASIS ACCIDENT CONDITIONS

In our letter dated January 28, 1997, AEP:NRC:1256A, we responded to generic letter (GL) 96-06. In that letter, we noted the following lines were potentially subject to thermally induced overpressurization, and further analyses were required to determine if the presently installed systems were acceptable or if modifications were required.

Reactor coolant pump seal water return line
Accumulator sample line
Reactor coolant system sample lines

Analyses of these lines have been completed. These analyses evaluated the pressure increase that would occur if the water in an isolated section of piping were heated from its normal operating temperature and pressure to the final containment temperature that exists following an accident. These analyses have shown that thermally induced pressurization of these lines during a design basis accident will not result in piping streses that exceed FSAR allowables.

A discussion of our findings is contained in the attachment to this letter.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

vlb

Attachment 270028

c: A. A. Blind
A. B. Beach
MDEQ - DW & RPD
NRC Resident Inspector
J. R. Padgett

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 20 DAY OF May 1997

Janice M. Bickers
Notary Public

My Commission Expires 2001

JANICE M. BICKERS
Notary Public, Berrien County, MI
My Commission Expires Feb. 16, 2001

A073/4

ATTACHMENT TO AEP:NRC:1256B

RESPONSE TO GENERIC LETTER 96-06
ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT
INTEGRITY DURING DESIGN BASIS ACCIDENT CONDITIONS

Introduction

Generic letter (GL) 96-06 requested licensees to determine:

1. if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions; and

2. if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

In our 120 day response to GL 96-06, AEP:NRC:1256A dated January 28, 1997, we showed that our containment air cooler cooling water systems are not susceptible to waterhammer or two-phase flow during postulated accident conditions. However, we did identify five fluid lines in both unit 1 and unit 2 where protection from thermal overpressurization was not readily identifiable. For those specific lines, we committed to performing additional analyses to determine whether their design is acceptable or warrants modifications.

As part of our 120 day response to GL 96-06, a review was completed to determine if the fluid lines that penetrate containment are susceptible to thermal overpressurization following a postulated accident. The review, focusing on both safety-related and non-safety-related systems, indicated that fluid lines penetrating containment fell into one of three categories: (1) those where overpressure protection is provided; (2) those where pressure relief can be predicted to keep the pipe stresses below the FSAR allowable stresses for emergency conditions; and (3) those where pressure relief cannot be predicted to limit piping stresses below FSAR allowable stresses for emergency conditions. Those lines that fell into categories (1) and (2) are within their design basis and, therefore, require no additional analysis. For those lines in category (3), pressure relief cannot be predicted to keep the lines within their design basis; therefore, we performed additional analyses. The results of those additional analyses are reported herein.

Susceptibility of Fluid Lines Penetrating Containment to Over Pressurization Due to Thermal Expansion

For those lines that did not fall into category (1) above, stress and force analyses were completed for the 120 day response to answer the question, "Is pressure relief available to prevent the stresses in the pipe wall from exceeding FSAR allowable stresses for emergency conditions?" Using that approach, all but five lines in each unit were eliminated from concern, those being the reactor coolant pump seal water return line, the accumulator sample line, and the three reactor coolant sample lines. For those five lines, enthalpy balances were completed to answer the question. Those results were used to draw the final conclusions about Cook Nuclear Plant's susceptibility to thermal overpressurization.

(1) Reactor Coolant Pump Seal Water Return Line

This 4" line returns reactor coolant pump seal leak-off to the chemical and volume control system for normal plant operation. The line has one motor-operated containment

isolation valve inside containment and a second motor-operated containment isolation valve in series outside containment. These gate valves close on a containment isolation signal, isolating the line passing through the containment penetration. The piping system design pressure at the penetration is 150 psig and no pressure relief mechanism is provided for the piping between the two containment isolation valves. Overpressure protection is provided for the piping system inside containment upstream of the inboard containment isolation valve via a safety valve. This configuration is similar for both unit 1 and unit 2.

Because pressure relief cannot be predicted for the section of pipe between the two containment isolation valves, an enthalpy balance was completed to determine the maximum internal pressure that would develop in the pipe following a postulated accident. That calculated pressure was then compared to the pressure required to induce wall stresses in excess of FSAR allowable stresses for emergency conditions. For an initial temperature and pressure of 150°F and 100 psia (measured conditions during normal operation with added conservatism) and a final containment temperature of 250°F (post-accident, the containment temperature reaches an initial peak, recovers to less than 250°F within a few minutes, then slowly declines), the pressure developed in the pipe was below the pressure at which the wall stresses would exceed FSAR allowable stresses for emergency conditions. Thus, the reactor coolant pump seal water return line is within its design basis and does not require modifications.

(2) Accumulator Sample Line

This 1/2" line is a common sample line from the four accumulator tanks. The line has two air-operated containment isolation valves in series outside containment that close on a containment isolation signal. Inside containment, there are normally closed air-operated globe valves located at each accumulator tank. The piping system design pressure is 600 psig and no pressure relief mechanism is provided between the normally closed accumulator sample valves inside containment and the containment isolation valves outside containment. This configuration is similar for both unit 1 and unit 2.

Because pressure relief cannot be predicted for the section of pipe between the accumulator sample valves inside containment and the containment isolation valves outside containment, a thermodynamic analysis was completed to determine the maximum internal pressure that would develop in the pipe following a postulated accident. That calculated pressure was then compared to the pressure required to induce wall stresses in excess of FSAR allowable stresses for emergency conditions. For an initial temperature and pressure of 70°F and 600 psia (measured conditions during normal operation with added conservatism) and a final containment temperature of 250°F, the pressure developed in the pipe was below the pressure at which the wall stresses would exceed FSAR allowable stresses for emergency conditions. Thus, the accumulator sample line is within its design basis and does not require modifications.

(3) Reactor Coolant System Sample Lines

These three 1/2" sample lines (pressurizer liquid space, pressurizer steam space, and hot leg samples) from the reactor coolant system all share a common configuration. Each line has two air-operated containment isolation valves in series outside containment. Inside containment, there are normally closed air-operated valves from each sample point. The piping system design pressure is 2485 psig and no pressure relief mechanism is provided between the sample valves inside containment and the containment isolation valves outside containment. This configuration is similar for both unit 1 and unit 2.

Because pressure relief cannot be predicted for the section of pipe between the sample valves inside containment and the containment isolation valves outside containment, a thermodynamic analysis was to be completed to determine the maximum internal pressure that would develop in the pipe following a postulated accident. However, when the operating conditions were reviewed to determine the initial conditions for the analysis, the operating temperatures of the sample lines were found to be greater than the post-accident containment temperature. Consequently, following a postulated accident, the sample lines will actually cool down instead of heat up.

These lines are not in continuous use. As a result, after the samples are pulled and the lines are isolated, the fluid cools to ambient containment temperature. As the fluid cools, its specific volume decreases resulting in decreased internal pressure. In the post-LOCA containment environment, the isolated sample lines would heat up to 250°F. This is less than the temperature of the fluid when it was isolated; thus, the post-LOCA internal pressure in the sample lines would remain below the design pressure for those lines. As such, the reactor coolant system sample lines are not susceptible to thermal overpressurization and do not require modification.

Summary

In our 120 day response to GL 96-06, we committed to performing additional analyses to address five fluid lines for which pressure relief could not be predicted. Those lines were the reactor coolant pump seal water return line, the accumulator sample line, and the three reactor coolant system sample lines. An enthalpy balance was completed for each of the aforementioned lines. In each case, we determined the line would not pressurize to the point where the wall stresses would exceed FSAR allowable stresses for emergency conditions. Thus, Cook Nuclear Plant is not susceptible to thermal overpressurization and no plant modifications will be made.