REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8005280598 DOC.DATE: 80/05/22 NOTARIZED: NO DOCKET #
FACIL:50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316
AUTH.NAME AUTHOR AFFILIATION
DOLAN,J.E. Indiana & Michigan Electric Co.
RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Requests that NRC allow return of steam generator level low-low setpoints to pre-790630 values.Factors inherent in revised setpoints lead to overly conservative setpoints, causing operational difficulties.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 4
TITLE: General Distribution for after Issuance of Operating Lic

NOTES: I&E - 3CYS ALL MATL.

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC ORB #1	7	7			
INTERNAL:	01 REG FILE	1	1	02 NRC PDR	1	1
	12 I&E	2	2	15 CORE PERF BR	1	1
	17 ENGR BR	1	1	18 REAC SFTY BR	1	1
	19 PLANT SYS BR	1	1	20 EEB	1	1
	21 EFLT TRT SYS	1	1	EPB-DOR	1	1
	OELD	1	0	STS GROUP LEADR	1	1
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	23 ACRS	16	16			

MAY 29 1980

MA
4

60

TOTAL NUMBER OF COPIES REQUIRED: LTTR ~~38~~ 41 ENCL ~~37~~ 40

INDIANA & MICHIGAN ELECTRIC COMPANY

P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

May 22, 1980
AEP:NRC:00271B

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

By letter dated June 21, 1979, AEP-79-16, the Westinghouse Electric Corporation (W) informed AEP of their intention to submit a 10 CFR 21 report to the Nuclear Regulatory Commission concerning heatup of the steam generator water level monitoring system reference leg, and of the possible non-conservative bias in indicated level, following a feedwater line rupture inside containment. The W 10 CFR 21 report was submitted to the NRC via Mr. T. M. Anderson's letter of June 22, 1979 to Mr. V. Stello (W letter No. TMA-2104). On August 13, 1979 the Commission issued IE Bulletin No. 79-21 subsequent to Staff review of the W report. Our response to the Bulletin, AEP:NRC:00271, was forwarded to the NRC on November 5, 1979. The "Steam Generator Level Low-Low" setpoints had been revised on June 30, 1979.

Recent discussions with members of your staff and their consultants have caused us to question the necessity of revising the "Steam Generator Level Low-Low" setpoints, given the overall conservatism of the W analysis. For reasons discussed below, I&MECo requests to be allowed to return the "Low-Low Level" setpoints to their values prior to the June 30, 1979 date ("the original setpoints").

The potential safety concern addressed in the W report involves the non-mechanistic rupture of a main feedwater pipe inside containment and the consequential heatup of the steam generator water level monitoring system reference leg. Increased reference leg water column temperature would result in a decrease in water column density and an apparent bias

A001/1

8005280598

in the indicated steam generator water level. According to W, this potential level bias could delay or prevent the primary signal arising from decreasing steam generator water level ('Steam Generator Water Level Low-Low').

As was recently discussed with members of your staff, there are a significant number of conservative factors inherent in the revised setpoints. The combination of these factors, which are discussed below, results in an overly conservative setpoint which, in turn, results in operational difficulties and unnecessary challenges to the Reactor Protection System/Engineered Safety Features Actuation System (RPS/ESFAS).

As stated in the W 10 CFR 21 report, the "Steam Generator Water Level Low-Low" function is the primary signal only for the feedwater line break scenario. Review of the analysis contained in Section 14.2.8 of Chapter 14 of the FSAR (Unit No. 2 'Yellow Pages') indicates that the following safeguards signals are also available to initiate a reactor trip and/or safeguards actuation for a feedline rupture downstream of the containment isolation check valve:

I. Reactor Trip

(a) Safety Injection Actuation

(b) Pressurizer Pressure-High

(c) Over Temperature Delta-T

(d) Steam Generator Water Level Low-Low (In Any Steam Generator)

(e) Steam Generator Water Level Low Coincident with Steam Flow/Feedwater Flow Mismatch (In Any Steam Generator)

II. Safety Injection Actuation

(1) Containment Pressure High (1.2 psig)

(2) - Steamline Pressure Low (Unit No. 2)

- Steam Flow High Coincident with Steamline Pressure Low or RCS T avg Low-Low (Unit No. 1)

(3) High Steamline Differential Pressure.

The aforementioned FSAR analysis conservatively assumes instantaneous loss of main feedwater flow to all four steam generators at the time of the break. Obviously, such an assumption is very conservative. Theoretically, there are three event scenarios which would result in the instantaneous termination of main feedwater flow to all four steam generators; (1) the simultaneous closure of the feedwater isolation valve on each secondary loop, (2) the simultaneous rupture of all four feedwater lines, and (3) the simultaneous and instantaneous loss of both feedwater pumps. Scenario (2) is an extremely low probability event well beyond the design basis of the Cook Plant. Scenarios (1) and (3) would automatically trip the turbine (through BOP circuitry) and initiate auto-start of the motor driven auxiliary feedwater pumps, regardless of steam generator water level. A turbine trip will initiate a reactor trip (through safety grade circuits) above 10% reactor power.

The steam generator level setpoints include an environmental allowance associated with the Barton Transmitters employed in these channels. This environmental allowance corresponds to the transmitter input/output characteristics under the severe environmental conditions of a LOCA and a steam line break. The environment which would exist following a hypothetical feedline rupture is much less severe than that to which the transmitters were tested. Thus, it would be reasonable to say that the environmentally induced transmitter error following a feedline break would be significantly less than that experienced during qualification testing. It should also be noted that the peak containment temperature following a feedline break would most probably be on the order of 250°F which is the saturation temperature at 10 psig. The 'Containment Pressure High' (Hi-1) setpoint which generates safety injection and reactor trip signals is 1.2 psig. Containment Pressure would exceed the 'Hi-1' setpoint almost immediately after initiation of the break. For conservatism, the 'Hi-1' functional channel was not assumed to be the principal trip/actuation function in the FSAR analysis.

Review of the FSAR steam line break analysis for Cook, which bounds the main feedwater line break in terms of containment response, shows that the containment temperature never exceeds 200°F prior to the 'Containment Pressure High' setpoint being reached, and prior to reactor trip, feedwater isolation, and auxiliary feedwater actuation being initiated. A containment temperature of 200°F corresponds to a correction of 4% in steam generator level, if the time required to heatup the reference leg is not taken into account. The temperature in the steam generator enclosures during normal power operation is on the order of 100-120°F. The relatively small temperature differential inside the enclosures for the accident (∠200°F @ 'Hi-1' setpoint) and normal operation (120°F) conditions coupled with the time dependent heat transfer rate between the containment atmosphere and the reference leg fluid provide reasonable assurance that the accuracy

of the steam generator water level indication system will not be adversely effected to any significant degree prior to reactor trip and safeguards initiation by the "Containment Pressure High" signal. This being the case, there would be no reason to incorporate a 'reference leg heatup' correction into the steam generator level setpoints.

Incorporation of the above discussed correction into the "Steam Generator Water Level Low-Low" setpoints has decreased the allowable operating range of steam generator water level. This results in operational difficulties during periods of low power and unit startup conditions and effectively increases the probability of placing a demand on the RPS/ESFAS.

For reasons cited above, IMECo requests that the NRC allow the return of the "Steam Generator Water Level Low-Low" setpoints to the pre-June 30, 1979 values; 11% and 15% of span for Unit Nos. 1 and 2 respectively. Our Technical Specification change request, Change No. 5 in our AEP:NRC:00313 submittal, would then be withdrawn.

Very truly yours,

John E. Dolan

John E. Dolan
Vice President

cc: R. C. Callen
G. Charnoff
R. S. Hunter
R. W. Jurgensen
D. V. Shaller - Bridgman