EG&G Idaho, Inc.

FORM EG&G-398
(Rev. 11-79)

INTERIM REPORT

Accession No. ______________

Report No. EGG-EA-5229

Contract Program or Project Title:

Electrical, Instrumentation and Control System Support

Subject of this Document:

The Effect of Containment Temperature on Liquid Level Measurements, Donald C. Cook Nuclear Station, Unit 2, Docket No. 50-316, TAC No. 12982

Type of Document:

Informal Report

Author(s):

A. C. Udy

Date of Document:

August 1980

Responsible NRC Individual and NRC Office or Division:

Paul C. Shemanski, Division of Licensing

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. DE-AC07-76ID01570
NRC FIN No. A6256

INTERIM REPORT

8009260084

P

TECHNICAL EVALUATION REPORT
THE EFFECT OF CONTAINMENT TEMPERATURE ON LIQUID LEVEL MEASUREMENTS

DONALD C. COOK NUCLEAR STATION, UNIT NOS. 1 AND 2

Docket Nos. 50-315 and 50-316

August 1980

A. C. Udy
Reliability and Statistics Branch
Engineering Analysis Division
EG&G Idaho, Inc.

TAC Nos. 12981 and 12982

ABSTRACT

As an indirect result of an assumed loss-of-coolant accident, there is an apparent increase in the indicated water level for those systems whose sensors and reference legs are exposed to the elevated containment temperature. This report examines the effected systems, the effect on the initiation of safety systems, and the effect on the information displayed to the reactor operator.

CONTENTS

TECHNICAL EVALUATION REPORT
THE EFFECT OF CONTAINMENT TEMPERATURE ON LIQUID LEVEL MEASUREMENTS

DONALD C. COOK NUCLEAR STATION, UNIT NOS. 1 AND 2

1.0 INTRODUCTION

Based on the information supplied by Indiana & Michigan Electric Company (I&MECo), this report addresses the effect of the containment temperature on the steam generator and pressurizer water level detectors.

In June 1979, the Power Systems Division of the Westinghouse Electric Corporation (W) notified the Nuclear Regulatory Commission (NRC)[1] and W utility customers of corrections that should be applied to indicated steam generator water level and associated low water level protection system setpoints and emergency operating procedures. The problem identified was that, as the temperature of the level measurement reference leg increased due to a high-energy line break, the water column density decreased. This appears as an apparent increase in the indicated water level, which could result in delayed protection (reactor trip and auxiliary feedwater) signals.

On August 13, 1979, the NRC sent an IE bulletin (#79-21)[2] to all reactor facilities. Boiling water reactors and those facilities with construction permits were notified for information. All operating pressurized water reactor licensees were directed to take steps to evaluate the problem, take corrective actions needed, and to notify the NRC of all actions taken as a result of the evaluation.

I&MECo responded in a letter of November 5, 1979[3] and requested technical specification changes to Tables 2.2-1 and 3.3-4 on Feburary 22, 1980[4]. This report is a technical evaluation of the material submitted by, and actions taken by, I&MECo for the Cook units.

2.0 EVALUATION OF THE COOK NUCLEAR STATION, UNITS 1 AND 2

2.1 Review Guidelines. IE Bulletin No. 79-21 provided the pressurized water power reactor licensees with the following NRC guidelines:

1. Guideline 1 - The licensee is to review all liquid level measurement systems within containment. If the signals are used to initiate safety actions or to provide post-accident monitoring information, a description of the system is to be submitted.

2. Guideline 2 - For those systems identified by guideline 1, the licensee is to evaluate the effect of post-accident temperature on the indicated water level (in comparison to the actual water level, including all sources of error).

3. Guideline 3 - The licensee is to provide a listing of all safety and control setpoints used with the level instrumentation, and verify proper setpoint actuation throughout the range of ambient temperature (including accident temperatures).

4. Guideline 4 - If a change of setpoints is necessary to ensure safe action, the licensee is to describe the corrective action and state when the action was taken.

5. Guideline 5 - The licensee is to ensure that the operators are instructed on the potential for, and the potential magnitude of, erroneous level signals. The completion date for procedure changes and operator training is to be identified.

2.2 Description of Liquid Level Measurement Systems. I&MECo has identified the steam generator narrow-range water level (SG level) and pressurizer level (PZR level) systems as inside containment, and used to initiate safety functions or to provide post-accident monitoring information.[3] They provided the following description:

"The SG level reference leg is a conventional condensing pot open column system contained entirely within the lower volume of the containment. The PZR level high side reference leg is a sealed bellows type filled with distilled water and is contained partially in the lower volume and partially in the upper volume of the containment. The PZR level low side reference leg is a conventional open column system entirely contained in the lower volume of the containment."[3]

The FSAR indicates that the SG low-low level trip, on two out of three coincidence in any one steam generator, will cause a reactor trip. High pressurizer level serves as a backup to high pressurizer pressure on two out of three coincidence, to cause a reactor trip; however, the FSAR does not take credit for this function[3]. Both the SG level and PZR level provide post-accident monitoring information in the control room. Table 7.5-2 of the FSAR shows that the PZR level is designed to operate for 1/2 hour after an accident. Section 7.5.2 of the FSAR indicates that pressurizer pressure and level are the only transmitters inside containment that are required to actuate the Engineered Safety Features.

As a result of the W evaluation, I&MECo raised, on June 30, 1979, the SG low-low setpoints to 17% and 21% of the narrow range instrument span for Units 1 and 2, respectively.[3] The increase was chosen as recommended by W, to correspond the temperature limit at which the containment pressure-high setpoint would cause a safety injection.[3] I&MECo dismissed level bias associated with high SG water level at elevated containment temperatures as insignificant.

2.3 Evaluation of Liquid Level Measurement Systems. Guideline 1 requested the licensee to review all liquid measurement systems within containment. I&MECo has provided this review and descriptions of the pressurizer level and steam generator level transmitters.[3]

Guideline 2 requested the licensee to compare indicated and actual water levels in a post-accident environment. Reference 3 provides the requested comparison, and states that reference leg boiling will not occur.

Guideline 3 requested that the setpoints used on safety and control circuits be identified with verification of proper setpoint actuation thoughout the temperature range. Guideline 4 requested documentation of any corrective action taken. The SG low-low setpoints were increased at both units. The revised setpoints of 17% and 21% of narrow range instrumentation (Units 1 and 2, respectively) were incorporated in June 1979[3]. The increased setpoint is based on the W recommendation and analysis, and on the containment temperature expected before the high containment pressure reaches its setpoint for safety injection (which will cause a reactor trip).

Since no credit is taken in the safety analysis for the pressurizer level high trip, no changes were made to this setpoint. The FSAR (Section 7.5.3) states that the reference leg of the pressurizer level transmitter will not exceed 140°F. A corresponding temperature for the Steam Generator level transmitters is not established by the FSAR.

The signals identified and setpoints revised for the temperature range specified by I&MECo are within the analyzed limits of the FSAR and redundant instrument trip signals.

To ensure that the pressurizer does not become water solid, or that the pressurizer heaters do not become uncovered, I&MECo has established procedural limits to the allowable pressurizer level that takes the temperature effect on the level sensor into account.[3] The actions taken and the information documented satisfies guidelines 3 and 4.

Guideline 5 requires any procedure changes and operator training needed as a result of IE Bulletin No. 79-21 be scheduled. Reference 3 indicates that operator training and needed procedure changes have been completed.

3.0 SUMMARY

The material submitted by I&MECo for this review has been evaluated to the guidelines of IE Bulletin No. 79-21. The changes in operator training and procedural changes concerning the steam generator level and the pressurizer level signals satisfy the guidelines of the IE bulletin.

Changes to the steam generator low-low level trip setpoints satisfy the guidelines of the IE bulletin and the Westinghouse recommendations. The pressurizer level high trip setpoint was not changed since this was not credited with a safety function in the FSAR. This is acceptable as the trip is within the analyzed limits.

The NRC should approve the proposed technical specification changes for the steam generator low-low level setpoints of 17% for Unit 1 and 21% for Unit 2, with allowable limits of 16% and 20%, respectively.

4.0 REFERENCES

1. W letter, T. M. Anderson, to U.S. NRC, Victor Stello, "Steam Generator Water Level," NS-TMA-2104, June 22, 1979.

2. NRC letter to all power reactor licensees and construction permit holders, "IE Bulletin No. 79-21," August 13, 1979.

3. I&MECo letter, R. S. Hunter, to U.S. NRC, James G. Keppler, "Response to IE Bulletin No. 79-21," AEP:NRC:00271, November 5, 1979.

4. I&MECo letter, John E. Dolan, to U.S. NRC, Harold R. Denton, AEP:NRC:00313, February 22, 1980.