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May 28, 1997

Docket Nos.: 50-348
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10 CFR 50.73

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
LER 96-007-01
IEEE-279 Requirements Not Met For Protection
Channel III Steam Generator Instrumentation

Ladies and Gentlemen:

Joseph M. Farley Nuclear Plant Licensee Event Report No. 96-007-01 is being submitted in accordance with 10 CFR 50.73(a)(2)(ii)(B). If you have any questions, please advise.

Respectfully submitted,

Dave Morey

MGE:96-07-r1.doc

Enclosure

cc: Mr. L. A. Reyes, Region II Administrator
Mr. J. I. Zimmerman, NRR Project Manager
Mr. T. M. Ross, Plant Senior Resident Inspector

IE22/1

NRC FORM 366 (4-95)

U.S.NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Joseph M. Farley Nuclear Plant - Unit 1	05000348	1 OF 5

TITLE (4)

IEEE-279 Requirements Not Met For Protection Channel III Steam Generator Instrumentation

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
11	07	96	96	007	01	05	28	97	J. M. Farley - Unit 2	05000364

OPERATING MODE (9): 1

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. D. Hill, General Manager - Nuclear Plant	334 899-5156

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-space typewritten lines) (16)

At 1720 on November 7, 1996, with Unit 1 in Mode 1 operating at 100% and Unit 2 defueled, it was determined that both Units had a design condition that was outside their respective design basis. Unit 1 and Unit 2 have not met the requirements of IEEE 279-1971 since removal of the low feedwater flow reactor trip function following the installation of a median signal selector (MSS) on the steam generator water level control (SGWLC) input signals. The MSS was implemented as part of a plant reliability design change to reduce the impact of a single SGWLC channel failure. This MSS was installed during the Unit 1 12th and Unit 2 10th refueling outages (spring of 1994 and spring of 1995 respectively). It was determined that the use of the MSS precluded the need for the low feedwater flow reactor trip. The low feedwater flow trip function had satisfied IEEE 279-1971 requirements. A postulated failure of a common tap for channel III steam flow and narrow range steam generator water level was not considered in the design which results in a control/protection scenario that does not conform to the requirements of IEEE 279-1971. This was due to personnel error in that the designers and reviewers failed to adequately consider the requirements of IEEE 279 -1971. Farley Nuclear Plant (FNP) will maintain the steam flow channel input to the SGWLC System selected to channel IV, except when maintenance or testing is required on that channel. When channel III is selected the associated steam generator water level bistables will be tripped within 6 hours consistent with the time frame of Technical Specifications. Long-term corrective actions will be implemented in 1998 to restore design to full compliance with IEEE 279-1971.

Plant and System Identification

Westinghouse -- Pressurized Water Reactor
Energy Industry Identification System codes are identified in the text as [XX].

Description of Event

Previous plant design recognized that each Unit was susceptible to a control/protection interaction on the steam generator water level instrumentation that necessitated design provisions to satisfy the requirements of IEEE 279-1971. For each steam generator, the water level input to protection channel III was selectable as an input to the steam generator water level control (SGWLC) System to develop the level error portion of the logic which contributed to the positioning of the feedwater regulating valves. If this channel were selected and subsequently failed high, the feedwater regulating valve to the associated steam generator would close. This situation coupled with a second failure of a steam generator water level protection channel (as required by IEEE 279-1971) would disable the low-low steam generator water level reactor trip. In the Reactor Protection System (RPS) logic scheme, the low feedwater flow coincident with low steam generator water level reactor trip was the selected method to protect against this postulated scenario and satisfy the IEEE 279-1971 requirements.

In an effort to improve plant reliability, design change packages (DCPs) 93-1-8538 and 93-2-8626 were prepared. The objective of the design was to eliminate each Unit's susceptibility to the failure of a single channel of the SGWLC System, while still ensuring conformance to IEEE 279-1971 requirements. The design incorporated the use of a median signal selector (MSS) [SEL] which automatically selected the median of the three level signals (isolated protection outputs) for a given steam generator and therefore prevented feedwater from being controlled based on a failed (high or low) level channel. Based on this design, it was determined that the low feedwater flow coincident with low steam generator level reactor trip was not needed as a back-up to the primary low-low steam generator level reactor trip in order to meet IEEE 279-1971 requirements. Following NRC approval for changes to each Unit's Technical Specifications for the deletion of the low feedwater flow reactor trip, the design changes were implemented in the spring of 1994 for Unit 1 and spring of 1995 for Unit 2.

At 1720 on November 7, 1996, with Unit 1 in Mode 1 operating at 100% and Unit 2 defueled, it was determined that the post design change configuration resulted in both Units being outside their respective design basis based on a review of FSAR sections 3.1.20 and 7.2.2.2.1. These sections

state that all requirements of IEEE 279-1971 are met. However, the design change to install the MSS had not considered the potential for a failure of the common steam flow/steam generator water level tap on each steam generator (channel III) coincident with the failure of an additional water level channel on that same steam generator. If the channel III common tap were to sever, the associated level channel would fail high and the steam flow channel would fail low. Assuming channel III were selected as the control signal input, the steam generator water level control (SGWLC) [LC] System would close the feedwater regulating valve due to the failed low steam flow channel. Since the SGWLC System level mismatch would not have sufficient time to overcome the flow mismatch during the transient, there is a high probability that a low-low steam generator level trip would be required. IEEE 279-1971 section 4.7.3 requires that a second random failure be considered in this situation. Since only three level channels are provided, a postulated failure of a second level channel in conjunction with the channel III level failure would result in not satisfying the required two-out-of-three coincidence in the low-low steam generator level reactor trip logic. Based on the above, it was concluded that the requirements of IEEE 279-1971 have not been met on either Unit since design change implementation that installed the MSS and removed the low feedwater flow reactor trip function. The above mentioned postulated failure of a common tap for channel III steam flow and narrow range steam generator water level, that was not considered in the design, results in a control/protection interaction scenario that does not conform to the requirements of IEEE 279-1971 when channel III is selected for steam flow input to the SGWLC System. FNP has determined that the steam flow channel input to the SGWLC System will remain selected to channel IV, which precludes the non-conforming control/protection interaction, except when maintenance or testing is required on that channel. A one hour non-emergency notification was made to the NRC at 1739 on November 7, 1996 for Unit 1 under the provisions of 10 CFR 50.72(b)(1)(ii)(B). Unit 2 was not included in the notification due to being defueled.

Cause of Event

Personnel Error in that the designers and reviewers who developed the design packages for the implementation of a median signal selector on the steam generator water level control input signals and deletion of the low feedwater flow reactor trip failed to adequately consider the requirements of IEEE 279-1971.

Safety Assessment

Although not in conformance with the provisions of IEEE 279-1971, the steam generator protection channels have been determined to be operable using the guidance of Generic Letter 91-18 because they are capable of performing their function as described in the current licensing basis.

A Probabilistic Risk Assessment (PRA) analysis shows that the loss of a common leg for channel III coincident with a failure of a second channel of steam generator water level is an insignificant contributor to a loss of main feedwater accident scenario.

Corrective Action

FNP has implemented administrative controls to ensure that the steam flow channel input to the SGWLC System will normally remain selected to channel IV, which precludes the control/protection interaction scenario that does not meet the requirements of IEEE 279-1971. Channel III will only be selected for short periods of time to accommodate maintenance and quarterly testing of channel IV steam pressure and steam flow instruments. When channel III is selected for any reason, the associated steam generator water level bistables will be tripped in the Solid State Protection System within 6 hours consistent with the time frame of Technical Specifications, unless channel III is de-selected prior to that time.

Westinghouse has initiated a Design Review to investigate and determine the cause of the failure of the supplied design to meet IEEE 279-1971 requirements. Appropriate corrective actions will be developed based on the results of the investigation.

As a short-term enhancement, design changes have been initiated to ensure that a power failure in the 7300 Process Control Cabinets will not result in the undetected selection of the channel III input to the SGWLC System. Under the new plant design, channel IV will be automatically selected on a loss of power. These design changes were implemented during the fall 1996 refueling outage for Unit 2 and during the spring 1997 refueling outage on Unit 1.

Modifications to the SGWLC System will be implemented prior to Mode 3 entry following the Unit 1 15th refueling outage (fall 1998) and the Unit 2 12th refueling outage (spring 1998). These modifications will restore the design to full compliance with IEEE 279-1971.

Additional Information

The following LER has been submitted on the subject of supplied design not adequately addressing applicable design requirements:

LER 94-005-00 (Shared) - Missile Protection for Condensate Storage Tanks.