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May 23, 1997

Docket Number: 50-348

10 CFR 50.73

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
Licensee Event Report Number 97-008-00
Outside Of Design Basis Due To RCS
Support Gaps Not Being Consistent With Design

Ladies and Gentlemen:

Attached is Farley Nuclear Plant Licensee Event Report No. 97-008-00. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(ii). If you have any questions, please advise.

Respectfully submitted,

Dave Morey

MJA:97-08cov.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 1/1

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)

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TITLE (4)
Outside of Design Basis Due to RCS Support Gaps Not Being Consistent With Design

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
04	23	97	97	008	00	05	23	97		

OPERATING MODE (9): 6

POWER LEVEL (10): 000

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
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (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)

YES (If yes, complete EXPECTED SUBMISSION DATE)   X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

On April 23, 1997 with Unit 1 in Mode 6 at approximately 80°F reactor coolant system (RCS) temperature and 0 psig RCS pressure, it was determined that a condition existed where Unit 1 may have operated outside the design basis of the plant. An evaluation concluded on May 16, 1997 that Unit 1 had been operated in a condition outside the design basis of the plant. Specifically, a condition of high stresses in the RCS piping and lateral supports existed during plant operation due to gaps associated with the 1C steam generator (SG) support bumpers not being consistent with design. The evaluation concluded that the high stresses that occurred during normal plant operation did not exceed design code requirements. However, these stresses concurrent with the postulated safe shutdown earthquake (SSE) would have exceeded the code allowable stresses.

During normal plant operation, higher than normal vibration was noted on the reactor coolant pump in Loop C. After the shutdown for the scheduled refueling outage, an investigation determined that hot gaps associated with the 1C SG support bumpers appeared to be too large on two of the bumpers and no gaps were present on two others. It was determined that interference between the RCS loop and the bumpers occurred during plant operation, resulting in a condition of high stresses in the RCS piping and the LS-12 support location.

This event was caused by cognitive personnel error during the initial installation of the support shims when the shims on three 1C SG support bumpers were not installed in accordance with the design. The existing installed shim configuration has been modified. Hot gaps in all loops will be verified to be consistent with the original design.

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TEXT CONTINUATION**

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TEXT (If more space is required, use additional NRC Form 366) (17)

Plant and System Identification

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

Description of Event

On April 23, 1997 with Unit 1 in Mode 6 at approximately 80°F reactor coolant system (RCS) temperature and 0 psig RCS pressure, it was determined that a condition existed where Unit 1 may have operated outside the design basis of the plant. An evaluation concluded on May 16, 1997 that Unit 1 had been operated in a condition outside the design basis of the plant. Specifically, a condition of high stresses in the RCS piping and lateral supports existed during plant operation due to gaps associated with the 1C steam generator (SG) support bumpers not being consistent with design. The evaluation concluded that the high stresses that occurred during normal plant operation did not exceed design code requirements. However, these stresses concurrent with the postulated safe shutdown earthquake (SSE) would have exceeded the code allowable stresses.

During normal plant operation, higher than normal vibration was noted on the reactor coolant pump (RCP) in Loop C. After the shutdown for the scheduled refueling outage in March 1997, a walk down was performed to determine if there were any obvious reasons for the higher than normal vibration. During the walk down while the plant was in a hot standby mode it was observed that the hot gaps associated with the 1C SG support bumpers appeared to be too large on two of the bumpers and no gaps were present on two others. It was determined that interference between the RCS loop and the bumpers occurred during the hot condition, resulting in a condition of high stresses in the RCS piping and the LS-12 support location. This condition had apparently existed since initial plant startup. Actions were undertaken to ascertain the impact on the reactor coolant loop C and associated support structures. In addition, actions were taken to determine the cause of the condition and return the structure to a condition such that the gaps are consistent with the original design parameters. Subsequently, as part of repair modifications, it was determined that shims were not installed during initial installation in accordance with the design. The shims for supports LS-12, LS-10, and LS-11 were installed at locations LS-10, LS-11, and LS-12, respectively.

Cause of Event

This event was caused by cognitive personnel error during the initial installation of the support shims. A review of the identification stampings placed on the shims during plant construction indicates the shims were not installed in accordance with the design. The shims for supports LS-12, LS-10, and LS-11 were

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installed at locations LS-10, LS-11, and LS-12, respectively. A contributing cause of this event was that inadequate inspections were performed following installation of the shims.

Safety Assessment

Inspections of critical areas were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code to identify any areas of degradation, damage, or distress that may have resulted from the high stresses. No abnormal indications were found. In addition, an evaluation of the potential stresses due to the thermal binding was conducted to determine whether the RCS and its supports had exceeded design code requirements. All stresses for the RCS were found to be within design code requirements for normal operation. The only support component that was found to be inconsistent with normal condition limits was LS-12, which is the SG lower lateral strut that was found to have a measured cold gap approximately 0.7 inches less than the design gap value (1.312 inches vice 1.997 inches). Analysis results concluded the loading applied to LS-12 was not high enough to exceed the pre-load of the LS-12 bolting, thus, no damage or deformation would have been expected, nor was any found during the inspection. LS-12 met its code faulted condition limits and thus in conjunction with the results of the visual inspection of the support components, an adequate basis has been demonstrated that strut LS-12 was not adversely affected and that its ability to withstand the full design basis loads per the original plant design has been maintained.

The measured cold gap data obtained on all three reactor coolant loops demonstrates that all the support shims on Loops A and B were fabricated and installed correctly. The cold gap data also shows that after approximately 30 heatup and cooldown cycles on Unit 1, the thermal growth of the reactor coolant loops has remained approximately the same. Although a reduced gap condition existed at support LS-12, the reactor coolant components have returned to approximately their original cold position, providing decisive evidence that the RCS piping, components, and supports have not experienced any damage or permanent deformation. In addition, inspections performed during the current refueling outage indicates the 1C reactor coolant pump has not sustained any detrimental effects as a result of the slightly elevated vibration.

No significant seismic event has ever occurred at the Farley site since startup of Unit 1. Although no detailed analysis has been performed, the likely effects that a seismic event may have on the RCS piping and equipment supports have been considered, relative to the as-found support condition (including the effects associated with the larger than expected gaps). Engineering judgment has concluded that although some plastic deformation may occur during the SSE, the functional capability of the RCS and auxiliary systems would be maintained and the plant would have remained in a safe condition.

In summary, although the Unit 1 RCS and supports were in a condition that resulted in high stresses, results of inspections, analysis, and engineering judgment have concluded that the health and safety of the public were not affected.

Corrective Action

The existing shim configuration on 1C SG has been modified. The existing shims on Loop C SG lower lateral support have been modified to accept adjustable shim packs, which will allow adjustment capability in setting the gaps to the design values. With the proper as-designed hot gaps at the lateral support locations, the RCS will be restored to its as-designed condition.

The hot gaps in all loops will be verified to be consistent with the original design.

Additional Information

Gap measurements taken in the cold shutdown condition also indicated a minor gap discrepancy on the 1B loop intermediate leg. An evaluation concluded the reduced gap did not result in a stress condition and the minor discrepancy was corrected.

A review of Unit 2 startup records concluded that adequate inspections were performed following installation of the shims in that hot gap measurements were verified.

The following LERs have been submitted on the subject of being outside of design basis.:

LER 97-007-00 (Shared) - Outside of Design Basis due to Degraded Cork Material

LER 96-006-00 (Unit 1) - Kaowool Fire Barrier Systems Not Installed Per Design Drawings

LER 96-007-00 (Shared) - IEEE-279 Requirements Not Met For Protection Channel III Steam Generator Instrumentation

LER 95-007-00 (Shared) - Control Room Pressurization Units Moisture Controllers Incapable of Performing Their Intended Function

LER 95-003-00 (Shared) - Potential For Loss of Engineered Safety Features Actuation Logic

LER 94-005-01 (Shared) - Missile Protection for Condensate Storage Tanks