ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8910120095 DOC.DATE: 89/10/06 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
ALEXICH,M.P. Indiana Michigan Power Co. (formerly Indiana & Michigan Ele
RECIP.NAME RECIPIENT AFFILIATION
DAVIS,A.B. Document Control Branch (Document Control Desk)

SUBJECT: Responds to NRC 890905 ltr re violations noted in Insp Repts 50-315/89-22 & 50-316/89-22.

DISTRIBUTION CODE: IE01D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 9
TITLE: General (50 Dkt)-Insp Rept/Notice of Violation Response

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-1 PD	1	1	GIITTER,J.	1	1
INTERNAL:	AEOD	1	1	AEOD/DEIIB	1	1
	AEOD/TPAD	1	1	DEDRO	1	1
	NRR SHANKMAN,S	1	1	NRR/DEST DIR	1	1
	NRR/DLPQ/PEB	1	1	NRR/DOEA DIR 11	1	1
	NRR/DREP/EPB 10	1	1	NRR/DREP/RPB 10	2	2
	NRR/PMAS/ILRB12	1	1	NUDOCS-ABSTRACT	1	1
	OE LIEBERMAN,J	1	1	OGC/HDS1	1	1
	REG FILE 02	1	1	RES MORISSEAU,D	1	1
	RGN3 FILE 01	1	1			
EXTERNAL:	LPDR	1	1	NRC PDR	1	1
	NSIC	1	1			

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 23 ENCL 23

R I D S / A D D S

I D S / A D D S

Indiana Michigan
Power Company
P.O. Box 16631
Columbus OH 43216

INDIANA
MICHIGAN
POWER

AEP:NRC:1090H

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
INSPECTION REPORTS 50-315/89022 (DRS) AND 50-316/89022 (DRS);
RESPONSE TO VIOLATIONS

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

Attn: A. B. Davis

October 6, 1989

Dear Mr. Davis

This letter is in response to H. J. Miller's letter dated September 5, 1989, which forwarded the report of the routine safety inspection conducted from July 10 through August 15, 1989, on activities at the Donald C. Cook Nuclear Plant Units 1 and 2. The Notice of Violation attached to Mr. Miller's letter identified two severity level IV violations in the areas of procedural compliance and design control. These violations are addressed in the attachment to this letter.

Mr. Miller's letter requested that we provide a discussion of our plan of action to ensure that fillet welds installed on socket fittings (socket welds) in small bore piping systems (less than 2 1/2 inches diameter) are of adequate size. This letter also requested a response to a perceived ineffectiveness of interim corrective actions implemented in association with our design control process. These issues are addressed below. Item 1 also constitutes our special report on socket weld inspections committed to in discussions with your staff.

1. <u>Verification of Socket Weld Size Acceptability</u>

   In a previous Region III inspection (Inspection Reports 50-315/88028 and 50-316/88032) deficiencies in the documentation of QC inspection and acceptance criteria for socket welds were identified. The noted deficiencies centered on the weld size specified for original design versus that specified for QC inspections. In addition, the

IE01

actual measured weld size was not consistently documented following QC inspections. Mr. Miller's letter requested that we provide a plan of action to ensure that adequate weld size was installed during previous modification work.

Cook Nuclear Plant QC/NDE personnel have performed an inspection of a sample of socket welds to permit an assessment of the acceptability of socket weld sizes as installed. This inspection was initiated prior to receipt of Mr. Miller's September 5 letter. The acceptance criteria are based on the ANSI B31.1 code for pressure piping, 1983 edition which specifies a socket weld leg size of greater than 1.09 times the nominal pipe wall thickness. This acceptance criteria was applied in anticipation that other ongoing AEPSC activities directed toward a reconciliation of our original design codes versus later code editions and code cases would lead to adoption of the B31.1-1983 criteria for socket weld sizes.

The socket weld inspection was performed on various, accessible small bore piping systems in both units. These systems were considered to be typical in general of small bore piping installed in the plant in that the socket welds in the inspected systems would have been installed using the same procedures applied to socket welds in other small bore piping throughout the plant. In Unit 1, a total of 60 socket welds were inspected. All inspected welds in Unit 1 were within the established acceptance criteria. In Unit 2, an initial sample of 60 welds were inspected, with 6 welds initially identified as falling below the acceptance criteria. Consequently, an expanded sample of 200 welds in Unit 2 were inspected with an additional 4 welds falling below the acceptance criteria.

In all 10 cases where initial data indicated a weld size less than the acceptance criteria, a system operability determination was made by plant personnel and confirmed by AEPSC design within 72 hours. All systems in which potentially undersized welds were identified were determined to be operable and no compromise of plant safety occurred. Subsequent investigations have found that 2 of the 6 initially identified non-conforming welds in Unit 2 were actually within the established acceptance criteria. Preliminary evaluation of the remaining 8 welds indicates that the degree to which they are undersized versus the B31.1-1983 requirement is small enough that the systems in which they are installed will continue to perform their design function and will not impact plant safety. The

limiting dimensions (leg and throat sizes) on all these welds were within 1/32 inch of the acceptance criteria.

All observed discrepancies have been included in our problem report system. Our evaluation of the root cause of the observed discrepancies and identification of any required corrective actions are continuing. No further inspections of socket welds are planned at this time. We are, however, continuing with more detailed evaluations of plant as-found data to confirm the conclusions given above. As previously agreed with Region III, we will continue to inform you of any new information arising from these evaluations. We are reviewing appropriate codes, code cases, etc. for applicability to the results of the socket weld inspections and are continuing the code reconciliation activities mentioned above. Any licensing or regulatory issues arising from these efforts will be addressed as required.

2. Design Control Corrective Actions

Our submittal of June 2, 1989, (AEP:NRC:1060N) provided a description of the actions that have been initiated in conjunction with other existing internal review programs to confirm the overall adequacy of our installed systems and design activities, as well as to expedite the implementation of any required improvements to our design control process. As discussed in the June 2 submittal, a management steering committee comprised of representatives from the AEPSC Design Division, Nuclear Engineering Department, Quality Assurance Division and Nuclear Safety and Licensing Section was established to direct the implementation of these activities. The programs initiated under the cognizance of the steering committee are intended to provide the insights necessary to establish a root cause for any identified deficiencies in our design and design control process in order to allow initiation of a definitive course of corrective action to eliminate the deficiency.

Several of the actions discussed in the above referenced June 2, 1989 response to the design control violation cited in Inspection Reports 50-315/88028 and 50-316/88032 have been initiated or completed in approximately the same time frame in which the inspection activities addressed by this response were conducted. Specifically, a review by an independent contractor of a sample of AEPSC design calculations was completed; a detailed AEPSC Quality Assurance Division audit

of installed, design complete and in-design requests for change (RFCs) and minor modifications (MMs) has been completed with the assistance of an independent contractor; walkdowns of safety related large bore (greater than 2 1/2 inches) piping systems and pipe supports have begun for the purpose of development of as-built drawings and independent analysis of as-built large bore piping systems for comparison to design analyses previously performed under AEPSC cognizance. Our assessment of the programs initiated to date is that they are well focused and should effectively lead to a determination of the root cause of any identified deficiencies. The evaluations of the results of completed activities, including final root cause determinations, are in progress. Until these evaluations are complete, a definitive statement as to the root cause of any identified design process weakness cannot be made. In the absence of a complete determination of the underlying reasons for weaknesses disclosed through our design confirmation programs, initiation of actions beyond those currently planned would be premature. Upon completion of the evaluations of the results of our ongoing design review activities an assessment will be made of the need for an expansion in the scope of our design confirmation program. In the interim, consistent with the commitment we have made to the Region III staff, we will continue to inform you of the progress of our design confirmation activities.

In the specific areas of piping and pipe support design, additional actions as described in the attached response to the NRC Notice of Violation have been initiated to improve design calculation documentation. It should also be noted that effective September 1, 1989 a reorganization in the Design Division occurred creating the Nuclear Design Group to provide focused management attention in the area of Cook Nuclear Plant and nuclear design activities.

This document has been prepared following Corporate procedures that incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

Sincerely,

M. P. Alexich
Vice President

eh

Attachment

cc: D. H. Williams, Jr.
A. A. Blind - Bridgman
R. C. Callen
G. Charnoff
NFEM Section Chief
A. B. Davis - Region III
NRC Resident Inspector - Bridgman

ATTACHMENT 1 TO AEP:NRC:1090H

RESPONSE TO VIOLATIONS

NRC Violation 1

"10 CFR 50, Appendix B, Criterion V, as implemented by the D. C. Cook Operations Quality Assurance Program requires that activities affecting quality shall be accomplished in accordance with prescribed procedures.

Contrary to the above, as required by D. C. Cook Procedure PMI-7030, Paragraph 4.5.1, a Condition Report was not initiated when design deficiencies were discovered for pipe supports 1-GFW-L812, 1-ACA-R911, 1-ACA-R910, 1-ACCW-R924, and 1-GCCW-L181 (315/89002-01; 316/89022-01).

This is a Severity Level IV violation (Supplement 1)."

Response to Violation

At the time the design discrepancies were identified, the primary effort was focused on completing the required piping system stress analysis, taking into consideration the additional identified support discrepancies, to complete the investigation and identification of required corrective action to allow re-start of the unit. Documentation of the discrepancies was not placed in the problem report/problem report trending system.

(1) Corrective Actions Taken and Results Achieved

Problem reports for the discrepant supports have been initiated for inclusion in the problem report/problem report trending system for Cook Nuclear Plant.

(2) Corrective Action Take to Avoid Further Violation

Pipe support discrepancies found during the establishment of the as-installed configuration of the piping system are factored into the pipe stress analysis. In order to ensure that these discrepancies are documented through our problem report system, the analysis verification checklist will be revised to include an explicit review prior to close-out of the analysis package to determine if the initiation of additional problem reports is required.

(3) Date When Full Compliance Will be Achieved

Problem reports for the discrepant pipe supports in the cited violation were initiated on August 25, 1989 (1-GCCW-L181), August 31, 1989 (1-GFW-L812) and October 4, 1989 (1-ACA-910 & 911 and 1-ACCW-R924). The revision to the analysis verification checklist will be completed by October 31, 1989.

NRC Violation 2

"10 CFR 50, Appendix B, Criterion III, as implemented by the D. C. Cook Operations Quality Assurance Program requires that measures be established to check and verify the design.

Contrary to the above, the design was not adequately checked and verified in that:

a. For pipe support 1-CCW-L181, the bending stress in the redesigned support was not evaluated (315/89022-03A; 316/89022-03A).

b. For pipe support 1-CS-R541, two of four load vectors on the support were not evaluated (315/89022-03B; 316/89022-03B).

This is a Severity Level IV violation (Supplement 1)."

Response to Violation:

a) During the calculation process the bending stress at the shear pin was judged to be acceptable, however, this engineering judgement was not documented in the calculation package. A confirmatory calculation was performed during the inspection showing that the bending stresses are within the design criteria limits.

b) During the calculation process, all four load conditions were considered with the thermal load determined to be the condition dictating which vectors were to be used for the calculation. The critical load vectors were subsequently used in the design, however, the calculation package did not document the engineering judgement that the other load vector combinations were non-critical.

(1) <u>Corrective Actions Taken and Results Achieved</u>

Preliminary confirmatory calculations were performed at the time of the inspection and indicated that the supports are within the design criteria limits.

(2) <u>Corrective Action Taken to Avoid Further Violation</u>

Standard calculation forms for pipe supports are being developed along with a revision to the checklists to be used during the checking process. The revised checklist and standardized forms will explicitly assure that assumptions and engineering judgements associated with pipe support calculations have been documented and verified.

(3) <u>Date When Full Compliance Will be Achieved</u>

The calculations will be updated to reflect the above concerns, standard calculation forms will be developed, and a checklist revision completed by October 31, 1989.