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

INDIANA
MICHIGAN
POWER

AEP:NRC:1184H
10 CFR 2.201

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NRC INSPECTION REPORTS NO. 50-315/93012 (DRS)
AND 50-316/93012 (DRS)
REPLY TO NOTICE OF VIOLATION

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

Attn: Mr. J. B. Martin

January 4, 1994

Dear Mr. Martin:

This letter is in response to a letter from G. E. Grant dated November 24, 1993, which forwarded a Notice of Violation associated with a System Based Instrumentation and Control Inspection conducted by Zelig Falevits and others of your office during August 17 through September 28, 1993. The violations are associated with errors in a setpoint calculation for refueling water storage tank level instrumentation and for the lack of an unreviewed safety question determination for Temporary Modification 2-93-015, which installed an I to I converter in the Unit 2 feedwater pump control circuitry.

Our reply to the Notice of Violation is contained in Attachment 1. We were also requested to respond to open and unresolved items from the inspection report. Our response is contained in Attachment 2. In addition, we were requested to address non-cited deficiencies associated with Temporary Modification 2-93-015. Our response to this request is contained in Attachment 3.

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath statement is attached.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

dr

Attachments

~~9401110333~~

JAN 1 0 1994

cc: A. A. Blind
G. Charnoff
T. E. Murley - NRC
NFEM Section Chief
NRC Resident Inspector
J. R. Padgett

ATTACHMENT 1 TO AEP:NRC:1184H

REPLY TO NOTICE OF VIOLATION

NRC Violation I (Severity Level IV)

"10 CFR 50, Appendix B, Criterion III, states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, drawings, procedures, and instructions and that the design control measures shall provide for verifying the adequacy of the design.

Contrary to the above, on September 1, 1993, the team noted that:

a. RWST instrumentation loop setpoint calculation No. 1-2-I9-03, dated August 25, 1993:

   (1) Erroneously derived the setpoint uncertainty value based on the use of Model N-E13 RWST transmitters. However, the installed RWST level transmitters were Model E13DM-HSAH1.

   (2) Provided no justification for using transmitter elevation 599'3" to derive the setpoint uncertainty value in the setpoint calculation.

   (3) Did not consider the error effects of the velocity head of the ECCS pump flow (Safety Injection and Residual Heat Removal pumps) during design basis accidents.

b. Flow diagram OP-1-5144-13, "Containment Spray System Unit #1," incorrectly identified RWST level transmitter ILS-950 as having a minimum level alarm at 638'11". However, the design basis setpoint value was 637' 2"."

Response to Violation I

1. <u>Admission or Denial of the Alleged Violation</u>

   Indiana Michigan Power admits to the violation as cited in the NRC Notice of Violation.

2. <u>Reasons for the Violation</u>

   The examples of the violation will be addressed individually.

   1.a.(1) The cause of the violation is attributed to lack of attention to detail in cross-checking the specific details in the documentation. The instrument data sheet used was from a slightly different model (N-E13 versus the correct model E13 of the same vendor). Both types of instruments are used in the plant and have nearly identical performance characteristics. The correct model data sheet was compared to the incorrect model data sheet and minor differences in some of the uncertainty terms were found which required the

calculation to be revised. However, no significant differences were found that changed the end results of the calculation adversely. Therefore this error did not have an adverse effect on safety.

1.a.(2) As noted in Revisions 2 and 4 of the calculation, the elevations used in the calculation were based on field walkdown measurements, which were taken in 1979. During the inspection, the elevations were remeasured but did not exactly match the previous field walkdown measurements. Because of the time that has elapsed, it was not possible to positively identify the reason for the discrepancy. It is noted, however, that the differences between the as-built measurements taken during the inspection and the measurements used in the calculation differed only slightly. The differences in elevation between the as-built heights and the heights used in the calculation varied between 1 inch and 2.5 inches, compared to an instrument span of 363 inches. The worst case error in the non-conservative direction (as-built lower than calculation) was 2.5 inches, which equates to an error of approximately 0.7%. In other words, the RWST indication and alarm will occur 0.7% lower than actual level. However, since there is considerable margin built into the alarm setpoints (approximately 21% for the low alarm and approximately 7% for the low-low alarm), this measurement error had no adverse effect on safety.

1.a.(3) The alarms and trips associated with the RWST level instrumentation include both high and low level type functions. There are two high level functions. The High Level Alarm is used to ensure the operators do not inadvertently overflow the RWST when filling, and a Minimum Level alarm is used to alert the operators that Tech Spec required minimum RWST volume requirements are being encroached. There are also two low level functions. The Low Level alarm is used to alert the operator that RWST inventories are approaching a low level at which the operator should begin to transfer to ECCS recirculation mode, and a Low-Low Level alarm and RHR pump trip is used to alert the operator that RWST inventory is depleted and to protect the RHR pump from damage due to low NPSH.

Because the RWST level transmitter is tapped off the ECCS suction line at the bottom of RWST tank, velocity head effects can be induced when the ECCS pumps are running. The high level functions are not affected by this because these functions are only used when the ECCS pumps are not running. The low level functions are affected because the ECCS pumps are running when they are required to function. However,

computation of the velocity head effect shows it will only affect the low level functions in a conservative manner. The velocity head effect induces a negative bias that results in a level indication that is lower than actual level. The effect on the low level alarms and RHR pump trip functions is that they will occur sooner and therefore does not jeopardize the RHR pump or plant safety.

Velocity head effect was not addressed due to unawareness of its influence on this application. Criteria for when this effect is to be considered were not included in the procedure used for the preparation of this calculation. It should be noted that most tank level instruments are tapped on the side of the tank and are therefore not affected by the velocity head effects of the tank suction line.

1.b. Setpoint information for the Cook Nuclear Plant is controlled through the Plant Setpoint Document, rather than through the flow prints. Because of this, there was no systematic process to have setpoints placed on flow prints, or to update the flow prints if the setpoint changed. The setpoints displayed on the drawings are for reference and are used for understanding the drawing only.

3. <u>Corrective Actions Taken and Results Achieved</u>

The examples of the violation will be addressed individually.

1.a.(1) The correct model data sheet has been compared with the incorrect model instrument data sheet and no significant differences were found. The calculation will be revised to incorporate the correct instrument data sheet information.

1.a.(2) The calculation has been revised to reflect the correct as-built elevations for the RWST level transmitters.

1.a.(3) The calculation has been revised to address the velocity head effects. Review of other tank level applications found the CST tank level to have a similar velocity head effect which was not addressed. The CST calculation will also be corrected.

1.b. Drawing OP-1-5144 and OP-2-5144 were revised to reflect the correct setpoints.

4. Actions Taken to Avoid Further Violations

As a general comment, it is noted that numerous instrument setpoint calculations are being updated as part of the Reactor Protection and Control Systems Upgrade Project which will be implemented during the 1994 refueling outages. The specific examples of the violation are addressed individually, below.

1.a.(1) Training sessions were held for Corporate I&C engineers which emphasize that self-checking and engineering reviews and verification are expected to be such that errors in instrument model numbers and similar documentation data is discovered and corrected prior to document issue.

1.a.(2) A sampling of safety-related instruments will be checked to verify correct as-built elevations are incorporated into setpoint calculations. The sampling will be completed by May 31, 1994. Further preventive measures will be established depending on the results of the sampling.

1.a.(3) The engineering guide governing setpoint calculations was revised on November 15, 1993, to require that process measurement effects, such as velocity head effects, are considered, as necessary, as part of the calculation preparation.

1.b. Since the intent of setpoint information on flow prints is only to help in the understanding of the drawing, a note will be added to all affected flow prints which states:

"Caution! The setpoints indicated are provided only to assist in understanding the drawing. Refer to appropriate setpoint control document for actual device setpoints." The notes will be added by July 6, 1994.

5. Date When Full Compliance will be Achieved

The examples of the violation will be addressed individually.

1.a.(1) Full compliance will be achieved by January 10, 1994, when the calculation is revised to reflect the correct model performance characteristics.

1.a.(2) Full compliance was achieved on November 5, 1993, when the calculation was revised to incorporate the as-built transmitter elevations.

1.a.(3) Full compliance was achieved on November 5, 1993, when the calculation was revised to include consideration of velocity head effects. The calculation for the CST will be revised by January 10, 1994, to correct the similar deficiency identified during the review of the RWST calculation.

1.b. Full compliance was achieved on September 15, 1993, when the affected Unit 1 and 2 OP drawings were revised.

NRC Violation II (Severity Level IV)

"10 CFR 50.59 states licensees may make changes to the facility as described in the safety analysis report without prior Commission approval unless the change involves an unreviewed safety question. A written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question is required.

Section 10.5.1.1 of the UFSAR states that the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators. In addition, Section 14.1.9 analyzed a loss of normal feedwater from pump failures which could result in a reduction of the secondary system to remove heat generated in the reactor core.

Contrary to the above, on April 7, 1993, the licensee failed to perform an evaluation to determine that changes made to the feedwater pump speed control system by temporary modification 2-93-015 did not involve an unreviewed safety question."

Response to Violation II

1. Admission or Denial of the Alleged Violation

   Indiana Michigan Power denies the violation as cited in the NRC Notice of Violation.

2. Reasons for Denial of the Violation

   At the Cook Nuclear Plant, temporary modifications undergo a screening to determine if an unreviewed safety question determination is required to be performed pursuant to 10 CFR 50.59. The process we use is based on the guidance of NSAC 125 (June 1989), entitled "Guidelines for 10 CFR 50.59 Safety Evaluations." This document was prepared jointly by the Nuclear Management and Resources Council (NUMARC) and the Nuclear Safety Analysis Center of the Electric Power Research Institute (EPRI).

   The inspection report (page 14) states:

   > "By failing to recognize that the speed control system was described in the UFSAR, the licensee concluded that 10 CFR 50.59 was not applicable, therefore, no safety evaluation was performed. The licensee's failure to perform a safety evaluation is considered to be a violation of 10 CFR 50.59."

We disagree with the statement that the speed control system is described in the UFSAR. The UFSAR (Section 10.5.1.1) specifically states that "the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators." There is no description of the circuitry provided in this statement.

NSAC 125 recognizes that changes made to the facility may implicitly impact the UFSAR. For these cases, NSAC 125 states that:

> "If the SSC (structure, system, or component) is part of a larger SSC described in the SAR and if the change affects the design, function, or method of performing the function of the larger SSC AS DESCRIBED IN THE SAR (emphasis added) then a safety evaluation is required."

As discussed in the UFSAR, the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators. The temporary modification only added a device to isolate noise in the speed control circuitry. Neither the function of the main feedwater pumps as described in the UFSAR (to provide feedwater flow to the steam generators) nor the method of performing the function as described in the UFSAR (with variable speed pumps) were impacted by the temporary modification. Additionally, since there was no description of the feedwater control system in the UFSAR, neither the design of the control system nor the design of the feedwater system as a whole as described in the UFSAR were impacted by the change. Based on these considerations, we conclude that no unreviewed safety question determination was required pursuant to 10 CFR 50.59.

As described in NSAC 125,

> "The purpose of 10 CFR 50.59 is to preserve the original licensing basis in the information submitted to the NRC as part of the application for an operating license and in the final safety evaluation report (SER) issued by the NRC staff. The NRC relies on this information to conclude that an operating license can be issued without undue risk to the health and safety of the public. This regulation allows the licensee to make changes without prior NRC approval while maintaining the licensing basis. It defines conditions that must be met in determining if prior regulatory review is needed."

The level of detail included in the UFSAR regarding the feedwater system is relatively small. This is commensurate with the fact that the system is non-safety related. The NRC staff review of the system during the original licensing of the plant would be expected to be of a different level of detail than for those systems which are safety related. Since the staff review did not rely on a detailed description of the speed

control system in order to conclude that the feedwater system was adequate from a safety perspective, it is not reasonable to conclude that 10 CFR 50.59 would be applicable to changes to the circuitry. It is noted that Section 14.1.9 of the UFSAR, entitled "Feedwater System Malfunctions," analyzes a complete loss of feedwater (due to no specific reason) and concludes that:

> "a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves."

It is also noted (as acknowledged in the inspection report, page 14) that failure of the I to I converter would have the same effect as the failure of the hand/auto station already in the circuit. In other words, the change did not introduce a new failure mode into the system.

ATTACHMENT 2 TO AEP:NRC:1184H

RESPONSE TO OPEN AND UNRESOLVED ITEMS

The cover letter for Inspection Report 50-315/316 93-012 (DRS) requested we provide a written response to open and unresolved items in the inspection report. There is only one item in this category, Unresolved Item 315/316 93012-04(DRS). This unresolved item involves a calculation performed to verify that an instrument sensing line associated with the Unit 2 auxiliary feedwater system was adequately supported. The inspection report states that:

> "The licensee was in the process of performing a calculation to determine whether the sensing line installation was adequate. Pending review of the calculation, this item is considered unresolved."

The subject calculation was completed following the inspection, and, at the inspector's request the information was mailed in an overnight package to NRC Region III on October 20, 1993. The information, documented in a memo from S. J. Jarrett/H. P. Damasco to M. S. Ackerman dated October 19, 1993, follows.

AMERICAN ELECTRIC POWER

Date October 19, 1993

Subject Cook Nuclear Plant
NRC SBICI Inspection Assistance
NEDS Review of As-Found Conditions

From S. J. Jarrett/H. P. Damasco

To M. S. Ackerman

As per your request, the Nuclear Design - Structural & Analytical Section (NEDS) has reviewed the as-found condition detected by an NRC inspector during a NRC SBICI Inspection. The 3/8"ϕ, 1/2"ϕ, and 3/4"ϕ downstream low pressure instrument lines associated with 2-FFS-257, and branch lines off the feed water header line 2-FW-48 were believed to have piping/tubing spans that exceeded Alternate Analysis Criteria.

Based on the above findings, Nuclear Engineering Site Design Section (NESD) performed a detailed walkdown of the piping system (see Attachment A). In order to satisfy the seismic overlap criteria, NESD included additional piping and supports beyond the area addressed by the NRC inspector. In so doing, they found that two support components in the overlap region of the continuation line were missing.

Nuclear Design - Mechanical Section (NEDM), performed a preliminary assessment of the piping system as found by the NRC inspector and concluded in their E-Mail, dated September 17, 1993 to Stan Farlow (see Attachment B), that the minor deviations in some support span lengths have an insignificant effect on the operability and/or the design basis of the instrument line.

NEDS performed the as-found and the as-designed DBE seismic evaluations on the piping system, in Calculation No. DC-D-02-MSC-36, by using walkdown information (see Attachment A), and Ebasco/P-Delta (E/PD) STRUDL integrated computer program. The piping stresses and the valve accelerations in these analyses were found to be well within the design basis criteria allowable limits. Attachments C through F highlight the very small stress interaction ratios, and vertical and horizontal valve acceleration ratios resulting from the E/PD STRUDL analyses. These analyses were performed to confirm the NEDM preliminary engineering review.

Although the piping system in the as-found condition meets the design basis criteria limits, the two missing components are being installed on the piping system to reflect similarity of pipe supports on identical piping systems in the vicinity, and further to conform with good engineering practice.

It is NEDS understanding that Job Order No. C19393 has been initiated by Plant Maintenance to install the two missing components. If you have any further questions please contact these writers at extension 3157.

S. J. Jarrett/H. P. Damasco

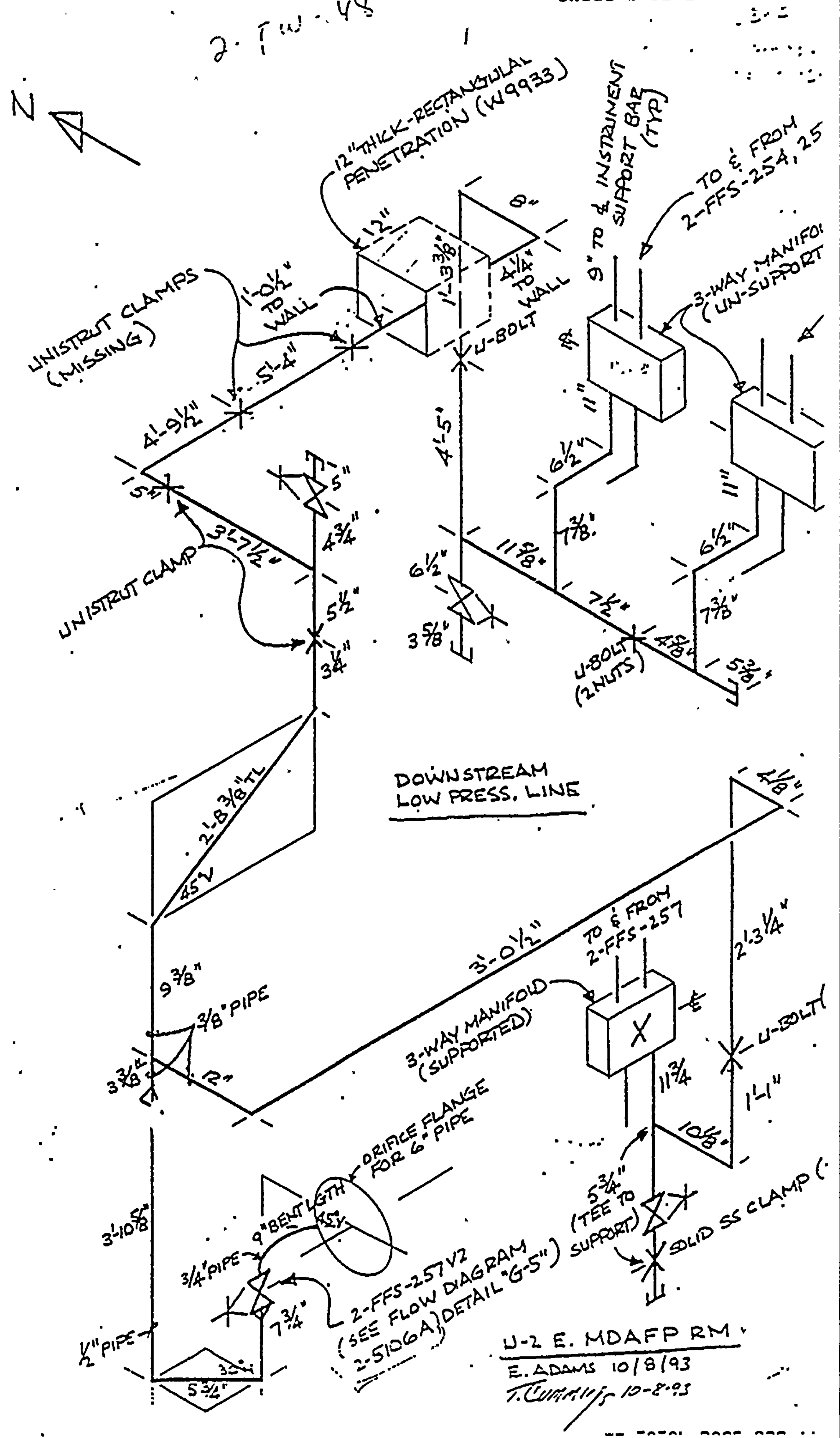
Approved by ______________________
FOR N. Ruccia, Manager
Nuclear Design - Structural & Analytical Section

/sjj

xc: R. C. Armstrong/S. P. Hodge - w/o attachment
A. K. Dey - w/o attachment
S. K. Farlow - w/o attachment
R. C. Carruth - w/o attachment
T. H. Cummings - w/o attachment

Attachment A

Copy of Fieldwalk from NESD


2-FW-48
N
12" THICK-RECTANGULAR PENETRATION (W9933)
9" TO ℄ INSTRUMENT SUPPORT BAR (TYP)
TO & FROM 2-FFS-254, 25
3-WAY MANIFOLD (UN-SUPPORT
UNISTRUT CLAMPS (MISSING)
1'-0½" TO WALL
4¼" TO WALL
12"
1'-3⅜"
8"
U-BOLT
5'-4"
4'-9½"
4'-5"
11"
6½"
5"
4¾"
3'-7½"
UNISTRUT CLAMP
5½"
3¼"
11⅝"
7⅞"
7½"
4⅝"
5⅜"
3⅝"
U-BOLT (2 NUTS)
DOWNSTREAM LOW PRESS. LINE
2'-8⅜" TL
45°
9⅜"
⅜" PIPE
3⅜"
12"
3'-0½"
TO & FROM 2-FFS-257
3-WAY MANIFOLD (SUPPORTED)
2'-3¼"
4⅛"
U-BOLT
11¾
1'-1"
10⅛"
ORIFICE FLANGE FOR 6" PIPE
9" BENT LGTH
45°
3'-10⅝"
¾" PIPE
½" PIPE
7¾"
2-FFS-257V2 (SEE FLOW DIAGRAM 2-5106A) DETAIL "G-5")
5¾" (TEE TO SUPPORT)
SOLID SS CLAMP
5¾"
U-2 E. MDAFP RM
E. ADAMS 10/8/93
T. Cummings 10-8-93

Attachment B

Copy of E-Mail to Stan Farlow (dated 9/17/93) from Amiya Dey

To: STAN K. FARLOW@NED@AEPSC
ROBERT C. CARRUTH@NED@AEPSC
Cc: Thomas H Cummings@SDO@COOK,STEVE P. HODGE@NED@AEPSC
Bcc:
From: AMIYA K. DEY@NDG@AEPSC
Subject: NRC Inquiry Relative To 2-FFS-257
Date: Friday, September 17, 1993 13:44:19 EDT
Attach:
Certify: N
Forwarded by:

Instrument Line 2-FFS-257

I understand that an NRC inspector has noted the piping line associated with the Instrument #2-FFS-257 to be under supported as compared to MDS-601 design standard.
Our findigs are given below:
The Standard MDS-601 was issued in Feb.1988 and I believe this instrument line was installed much earlier.This system and other instrument piping/tubing systems were installed per EDS Corp. developed guideline titled"Field Fabricated Support Guide For Small Diameter Pipes and Copper Tubing".This guideline was included as a section in a larger guideline known as the Alternate Analysis Criteria(AAC).
The AAC provides cookbook type directions in supporting piping/tubing systems and is very conservative.Our experience, with piping/tubing systems analysed and supported per AAC,indicates that:

*these systems generally are well designed and meet applicable code allowables easily when computer analysis is performed to verify system adequacy where limitations of the AAC have been marginally exceeded.

The instrument line being reviewed is 1/2 in. sch.80 pipe upto a TEE connection and then reduces to 3/8 in.sch. pipe sections.Our walkdown review of this systems indicates minor deviations in some span lenghths which we believe have no significance on the operability and/or the design basis of this insturment line.
If you need additional information,please contact me at 3860.
Thanks,
☺ Amiya ☺

Attachment C

Methodology

Method used to analyze the piping system in Calculation No. DC-D-02-MSC-36

AEPSC
DONALD C. COOK NUCLEAR PLANT , UNIT 2

6.0 METHODOLOGY

The piping system has been rigorously analyzed using walkdown information and Ebasco/P-Delta (E/PD) STRUDL integrated computer program. Additional hand calculations, where necessary, have been performed to develop problem input data and/or to justify acceptance of the piping system.
The information relevant to the final output is given below:

COMPUTER PROGRAM & VERSION:

E/PD STRUDL Version 0193 VAX VMS 5.5

JOB No.	RUN DATE & TIME	COMMENTS
2448	10-18-93 08:21	AF run - see comm. below
2449	10-18-93 08:23	DSN run - see comm. below

ADDITIONAL COMMENTS:

Job no. 2448 is the run simulating the as-found condition with Design Basis Earthquake response spectra.

Job no. 2449 is the as-design run that includes the two missing pipe clamps, and using the Design Basis Earthquake response spectra.

REV.	BY	DATE	CHK.	DATE
0	[signature]	10/18/93	[signature]	10/18/93

NUCLEAR DESIGN GROUP

CALCULATION No. DC-D-02-MSC-36

Attachment D

Problem Description

AEPSC
DONALD C. COOK NUCLEAR PLANT , UNIT 2

7.0 PROBLEM DESCRIPTION

GENERAL DESCRIPTION:

Analysis on the as-found 3/8", 1/2", 3/4" downstream low pressure instrument header and branch lines off feedwater line 2-FW-48 to address overspan concerns raised by an NRC inspector.

NUMBER OF SUPPORTS:

8 (including the two missing unistrut clamps)

PRIMARY BUILDING:

Turbine - Area 2T4

ELEVATION OF MAIN PIPING:

593'-0"

MAJOR EQUIPMENT:

None

0	[illegible signature]	10/18/93	[illegible initials]	10/18/93
REV.	BY	DATE	CHK.	DATE

NUCLEAR DESIGN GROUP

CALCULATION No. DC-D-02-MSC-36

Attachment E

This attachment shows stress ratios, and valve acceleration ratios, from the as-found analysis which excludes the two missing support components, that are well within the design basis criteria allowable limits.

Sheet 1	Stress Interaction Ratio
Sheets 2&3	Horizontal Valve Acceleration Ratios
Sheets 4&5	Vertical Valve Acceleration Ratios

```
//-PIPING-/-TBLNAM-/--CODE--/--------/--NODE--/EQUATION/------APPLIED FORCES-----/--------STRESS DATA---------//--LOAD--/-RESULT-//
   MEMBER   SECNAM  VERSION                               FX/MX    FY/MY    FZ/MZ    PRES/NONP   TOT/ALW          RATIO

                                                31.1-4      0.0      0.0      0.0      826.22    7335.86                      PASS
                                                            0.0      0.0      0.0     6509.64   37500.00           0.20
```

SUMMARY OF THE PIPING MEMBER CHECKS BY EQUATION

EQUATION	MAXIMUM IR	MEMBER	LOAD
31.1-1	0.4748	98	13
31.1-2NU	0.5887	44	1000
31.1-2E	0.5290	44	2000
31.1-3	0.0979	30	14
31.1-4	0.0979	98	

INTERACTION RATIOS

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 54 PIPE MEMBERS, THAT ARE CHECKED, PASSED CODE CHECKS.

```
$CHECK PIPE BREAK LOC ALL
$CHECK PIPE CRACK LOC ALL
$CHECK PIPE BRANCH DISPL ALL
 PRINT PIPE ADD STRESS INT FACT
```

Attachment E
Sheet 1 of 5

****
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE 2, RESULTS *
****

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA AX/MX	AY/MY	----------/ AZ/MZ	----ALLOWABLE DATA ACC	----/ MOM	--RATIO-/	-RESULT-// EQUATION
308		GATE1	123	308L	0.00	27.39	0.00	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			123	308M	0.00	27.39	0.00	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			456	308L	0.01	54.24	0.01	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE
			456	308M	0.01	54.24	0.01	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE
508		GATE3	123	508L	0.00	27.39	0.03	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			123	508M	0.00	27.39	0.03	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			456	508L	0.01	54.24	0.05	772.80	1000000.06	0.07	PASS
					0.00	0.01	0.00				ONE
			456	508M	0.01	54.24	0.05	772.80	1000000.06	0.07	PASS
					0.00	0.01	0.00				ONE
608		GATE2	123	608L	0.00	49.27	0.00	772.80	1000000.06	0.06	PASS
					0.00	0.00	0.00				ONE

Attachment E
Sheet 2 of 5

```
//-VALVE--/-TBLNAM-/------TITLE------/CRITICAL/-MEMBER-/----------APPLIED DATA----------/----ALLOWABLE DATA----/--RATIO-/-RESULT-//
   NODE     SECNAM                       LOAD                   AX/MX      AY/MY      AZ/MZ        ACC        MOM                 EQUATION

                                          123     608M          0.00      49.27       0.00      772.80 1000000.06      0.06       PASS
                                                                 0.00       0.00       0.00                                        ONE

                                          456     608L          0.01      97.46       0.01      772.80 1000000.06      0.13       PASS
                                                                 0.00       0.00       0.00                                        ONE

                                          456     608M          0.01      97.46       0.01      772.80 1000000.06      0.13       PASS
                                                                 0.00       0.00       0.00                                        ONE

  438              Y075T38                123     438L          0.00      27.32       0.00      772.80 1000000.06      0.04       PASS
                                                                 0.01       0.02       0.02                                        ONE

                                          123     438M          0.00      27.32       0.00      772.80 1000000.06      0.04       PASS
                                                                 0.01       0.02       0.02                                        ONE

                                          456     438L          0.01      54.06       0.01      772.80 1000000.06      0.07       PASS
                                                                 0.02       0.04       0.04                                        ONE

                                          456     438M          0.01      54.06       0.01      772.80 1000000.06      0.07       PASS
                                                                 0.02       0.04       0.04                                        ONE

   ********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

             ALL      4 PIPING VALVE    JOINTS  , THAT ARE CHECKED, PASSED CODE CHECKS.
$
$CHECK VALVE HORIZONT. ACCEL. FOR 3G
$
PIPE VALVE CHECK PARA
AXPFF 1.0000 LOADS ALL
AYPFF 0.0001 LOADS ALL
AZPFF 1.0000 LOADS ALL
AAF   1.5000 JOI   ALL
TRACE 2
END
LOAD LIST  1000  2000
CHECK PIPE VALVE INT EQU JOI -
   308        508        608        438 -
 MOCH TWO
```

Attachment E
Sheet 3 of 5

```
*********************************************************
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE  2, RESULTS *
*********************************************************
```

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA AX/MX	AY/MY	----------/ AZ/MZ	----ALLOWABLE DATA ACC	----/ MOM	-RATIO-/	-RESULT-// EQUATION
308		GATE1	1000	308L	41.19	0.00	41.10	1159.20	1000000.06	0.05	PASS
					0.00	0.01	0.00				ONE
			1000	308M	41.19	0.00	41.10	1159.20	1000000.06	0.05	PASS
					0.00	0.01	0.00				ONE
			2000	308L	81.58	0.01	81.39	1159.20	1000000.06	0.10	PASS
					0.00	0.01	0.00				ONE
			2000	308M	81.58	0.01	81.39	1159.20	1000000.06	0.10	PASS
					0.00	0.01	0.00				ONE
508		GATE3	1000	508L	40.48	0.00	259.97	1159.20	1000000.06	0.23	PASS
					0.00	0.00	0.00				ONE
			1000	508M	40.48	0.00	259.97	1159.20	1000000.06	0.23	PASS
					0.00	0.00	0.00				ONE
			2000	508L	80.16	0.01	524.42	1159.20	1000000.06	0.46	PASS
					0.00	0.01	0.00				ONE
			2000	508M	80.16	0.01	524.42	1159.20	1000000.06	0.46	PASS
					0.00	0.01	0.00				ONE
608		GATE2	1000	608L	40.89	0.00	46.15	1159.20	1000000.06	0.05	PASS
					0.00	0.00	0.00				ONE

Attachment E
Sheet 4 of 5

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA AX/MX	AY/MY	AZ/MZ----------/	----ALLOWABLE DATA ACC	MOM----/	--RATIO-/	-RESULT-// EQUATION
			1000	608M	40.89 0.00	0.00 0.00	46.15 0.00	1159.20	1000000.06	0.05	PASS ONE
			2000	608L	80.98 0.00	0.01 0.00	90.76 0.00	1159.20	1000000.06	0.10	PASS ONE
			2000	608M	80.98 0.00	0.01 0.00	90.76 0.00	1159.20	1000000.06	0.10	PASS ONE
438	Y075T38		1000	438L	39.75 0.04	0.00 0.12	45.44 0.07	1159.20	1000000.06	0.05	PASS ONE
			1000	438M	39.75 0.04	0.00 0.12	45.44 0.07	1159.20	1000000.06	0.05	PASS ONE
			2000	438L	78.19 0.05	0.01 0.14	89.34 0.09	1159.20	1000000.06	0.10	PASS ONE
			2000	438M	78.19 0.05	0.01 0.14	89.34 0.09	1159.20	1000000.06	0.10	PASS ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
PIPING LOAD TYPES ALEVEL     11   701 -
BLEVEL   1000 TO   1001 CLEVEL   2000 TO   2001 DEAD 11 -
THERMAL      1
 SECTION FR  NS 2  0.0  1.0
 OUTPUT DECIMAL 5
 OUTPUT BY MEMBERS
 LOAD LIST     11     1 701 1000 TO 1001 2000 TO 2001
$
UNIT INCH POUND RAD
LIST DISPLACEMENTS '&PIPEJOI'
```

Attachment E
Sheet 5 of 5

Attachment F

This attachment shows stress ratios, and valve acceleration ratios, from the <u>as-design analysis</u> which includes the two missing support components, that are well within the design basis criteria allowable limits.

Sheet 1	Stress Interaction Ratio
Sheets 2&3	Horizontal Valve Acceleration Ratios
Sheets 4&5	Vertical Valve Acceleration Ratios

//-PIPING-/-TBLNAM-/--CODE--/ MEMBER SECNAM VERSION	--------/--NODE--/EQUATION/	------APPLIED FORCES----- FX/MX	FY/MY	FZ/MZ	/--------STRESS DATA-------- PRES/NONP	TOT/ALW	//--LOAD--/ RATIO	-RESULT-//
	31.1-4	0.0	0.0	0.0	826.22	7368.35		PASS
		0.0	0.0	0.0	6542.13	37500.00	0.20	

SUMMARY OF THE PIPING MEMBER CHECKS BY EQUATION

EQUATION	MAXIMUM IR	MEMBER	LOAD
31.1-1	0.4766	98	13
31.1-2NU	0.4753	98	1000
31.1-2E	0.4770	44	2000
31.1-3	0.0973	30	14
31.1-4	0.0973	98	

INTERACTION RATIOS

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 54 PIPE MEMBERS, THAT ARE CHECKED, PASSED CODE CHECKS.

$CHECK PIPE BREAK LOC ALL
$CHECK PIPE CRACK LOC ALL
$CHECK PIPE BRANCH DISPL ALL
PRINT PIPE ADD STRESS INT FACT

Attachment F
Sheet 1 of 5

```
***********************************************************
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE  2, RESULTS *
***********************************************************
```

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/-TBLNAM-/------TITLE------/CRITICAL/-MEMBER-/----------APPLIED DATA----------/----ALLOWABLE DATA----/--RATIO-/-RESULT-//

VALVE NODE	TBLNAM SECNAM	TITLE	CRITICAL LOAD	MEMBER	AX/MX	AY/MY	AZ/MZ	ACC	MOM	RATIO	RESULT EQUATION
308		GATE1	123	308L	0.00	27.39	0.00	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			123	308H	0.00	27.39	0.00	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			456	308L	0.01	54.24	0.01	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE
			456	308H	0.01	54.24	0.01	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE
508		GATE3	123	508L	0.00	27.39	0.03	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			123	508H	0.00	27.39	0.03	772.80	1000000.06	0.04	PASS
					0.00	0.00	0.00				ONE
			456	508L	0.01	54.24	0.06	772.80	1000000.06	0.07	PASS
					0.00	0.01	0.00				ONE
			456	508H	0.01	54.24	0.06	772.80	1000000.06	0.07	PASS
					0.00	0.01	0.00				ONE
608		GATE2	123	608L	0.00	51.23	0.00	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE

Attachment F.
Sheet 2 of 5

//-VALVE-- NODE	/-TBLNAM-/ SECNAM	------TITLE------	/CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA---------- AX/MX	AY/MY	AZ/MZ	/----ALLOWABLE DATA---- ACC	MOM	/--RATIO-/	-RESULT-// EQUATION
			123	608M	0.00	51.23	0.00	772.80	1000000.06	0.07	PASS
					0.00	0.00	0.00				ONE
			456	608L	0.01	101.30	0.01	772.80	1000000.06	0.13	PASS
					0.00	0.00	0.00				ONE
			456	608M	0.01	101.30	0.01	772.80	1000000.06	0.13	PASS
					0.00	0.00	0.00				ONE
438		Y075T38	123	438L	0.00	27.64	0.00	772.80	1000000.06	0.04	PASS
					0.01	0.02	0.02				ONE
			123	438M	0.00	27.64	0.00	772.80	1000000.06	0.04	PASS
					0.01	0.02	0.02				ONE
			456	438L	0.01	54.70	0.01	772.80	1000000.06	0.07	PASS
					0.02	0.04	0.04				ONE
			456	438M	0.01	54.70	0.01	772.80	1000000.06	0.07	PASS
					0.02	0.04	0.04				ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
$
$CHECK VALVE HORIZONT. ACCEL. FOR 3G
$
PIPE VALVE CHECK PARA
AXPFF 1.0000 LOADS ALL
AYPFF 0.0001 LOADS ALL
AZPFF 1.0000 LOADS ALL
AAF   1.5000 JOI   ALL
TRACE 2
END
LOAD LIST  1000  2000
CHECK PIPE VALVE INT EQU JOI -
   308        508        608        438 -
 NOCH TWO
```

Attachment
Sheet 3 of 5

```
*********************************************************
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE  2, RESULTS *
*********************************************************

JOB ID -   FFS257       JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -  LENGTH     WEIGHT     ANGLE    TEMPERATURE    TIME      MASS
                INCH       KIPF       RAD      FAH            SEC       LBM
```

//-VALVE--/-TBLNAM-/	------TITLE------/	CRITICAL/	-MEMBER-/	----------APPLIED DATA		----------/	----ALLOWABLE DATA	----/	--RATIO-/	-RESULT-//
NODE SECNAM		LOAD		AX/MX	AY/MY	AZ/MZ	ACC	MOM		EQUATION
308	GATE1	1000	308L	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
		1000	308M	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
		2000	308L	81.56 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
		2000	308M	81.56 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
508	GATE3	1000	508L	40.63 0.00	0.00 0.00	308.47 0.00	1159.20	1000000.06	0.27	PASS ONE
		1000	508M	40.63 0.00	0.00 0.00	308.47 0.00	1159.20	1000000.06	0.27	PASS ONE
		2000	508L	80.47 0.00	0.01 0.01	620.58 0.00	1159.20	1000000.06	0.54	PASS ONE
		2000	508M	80.47 0.00	0.01 0.01	620.58 0.00	1159.20	1000000.06	0.54	PASS ONE
608	GATE2	1000	608L	40.76 0.00	0.01 0.00	48.66 0.00	1159.20	1000000.06	0.05	PASS ONE

//-VALVE--/-TBLNAM-/------TITLE------/CRITICAL/-MEMBER-/----------APPLIED DATA----------/----ALLOWABLE DATA----/--RATIO-/-RESULT-//

NODE	SECNAM	TITLE	LOAD	MEMBER	AX/MX	AY/MY	AZ/MZ	ACC	MOM	RATIO	EQUATION
			1000	608M	40.76	0.01	48.66	1159.20	1000000.06	0.05	PASS
					0.00	0.00	0.00				ONE
			2000	608L	80.72	0.01	95.58	1159.20	1000000.06	0.11	PASS
					0.00	0.00	0.00				ONE
			2000	608M	80.72	0.01	95.58	1159.20	1000000.06	0.11	PASS
					0.00	0.00	0.00				ONE
438		Y075T38	1000	438L	41.17	0.00	45.88	1159.20	1000000.06	0.05	PASS
					0.03	0.11	0.07				ONE
			1000	438M	41.17	0.00	45.88	1159.20	1000000.06	0.05	PASS
					0.03	0.11	0.07				ONE
			2000	438L	80.98	0.01	90.20	1159.20	1000000.06	0.10	PASS
					0.05	0.13	0.09				ONE
			2000	438M	80.98	0.01	90.20	1159.20	1000000.06	0.10	PASS
					0.05	0.13	0.09				ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
PIPING LOAD TYPES ALEVEL      11    701 -
BLEVEL   1000 TO   1001 CLEVEL   2000 TO   2001 DEAD 11 -
THERMAL      1
 SECTION FR  MS 2  0.0  1.0
 OUTPUT DECIMAL 5
 OUTPUT BY MEMBERS
 LOAD LIST      11      1 701 1000 TO 1001 2000 TO 2001
$
UNIT INCH POUND RAD
LIST DISPLACEMENTS '&PIPEJOI'
```

Sheet 5 of 5

ATTACHMENT 3 TO AEP:NRC:1184H

INFORMATION REGARDING TEMPORARY MODIFICATION

2-93-015

As requested in the cover letter for inspection report 50-315/316 93012 (DRS), this attachment provides the results of our investigation into the issues associated with Temporary Modification (TM) 2-93-015. This TM installed current-to-current (I/I) converter modules in the control circuitry for the Unit 2 East and West Main Feedwater Pumps. The purpose of the temporary modification was to isolate the control circuitry from the presence of signal grounds in the field wiring that were interfering with proper operation of the D/P slave controller. The modification was processed as a temporary modification because the source of the grounding problem could not be resolved with the unit on line. In addition, the control room portion of the feedwater control system circuitry has been scheduled for replacement in 1994 as part of the reactor protection system upgrade project.

Two design errors occurred that were found during the post-installation checkout process. Specifically, the design as released by Plant Engineering had the I/I wired in backwards (i.e., input and output reversed), and a necessary 100 ohm input resistor omitted. These errors were discovered during post-installation circuit checks, prior to post modification testing (PMT) and prior to placing the affected circuitry back into operation. Since the errors were identified and corrected prior to PMT, no condition report was required. Although not required, a condition report was subsequently generated to document our investigation.

The TM did not involve safety related equipment, and therefore the quality assurance requirements of 10 CFR 50 Appendix B and the requirements of ANSI-N45.2.11 do not specifically apply. However, since the temp mod process used at the plant is common to both safety-related and non-safety related changes, our review of this event was conducted to consider its implications for safety-related temp mods.

The following conclusions were drawn from our investigation of this event:

1. The design error occurred as the result of attempting to copy the design details of an existing I/I circuit from another approved plant drawing to the drawing markup which was used to serve as the installation drawing for the temp mod. While possible to correctly identify the input and output leads on this drawing, the input/output lead designations (W,X & D,F) were inadvertently transposed during their transfer to the mark-up. The 100 ohm input resistor was overlooked and was not transferred to the mark-up.

2. A vendor document detailing the design and installation requirements for the I/I was available, but not as an approved plant document. The engineer chose not to use it, because he felt confident that he was adequately familiar with the hardware and that reference to the vendor information would be unnecessary. Use of the document would likely have allowed the engineer to detect the transposition error, as well as the omission of the input resistor.

3. The engineer assigned to develop and implement the temp mod request was adequately qualified to perform this task.

4. The supervisor who reviewed the marked-up drawings was also adequately qualified to perform this function. However, the supervisor became involved with the development of the drawing markups. As a result, his review was not independent, which may have compromised his ability to detect the design errors.

5. The temp mod procedure does not clearly convey design review requirements. The Engineering Supervisor initialed the drawings in the temp mod package indicating his concurrence. However, this review is not a step required by the temp mod procedure.

6. The Plant Engineering review required by the temp mod procedure calls for "...a review of the request to ensure that it is needed, correct, practical, and that it accomplishes the intended purpose." It does not require a technical verification of the design change documents (i.e. drawing mark-up in this case) that are used to implement the request.

This event has identified a weakness in the temp mod procedure design review requirements. In order to prevent recurrence of an incident of this nature, the TM procedure, PMP.5040.MOD.001, will be revised to strengthen the requirements for verifying the technical accuracy of the design. The revision will clearly identify expectations and responsibilities for the review. These revisions will be completed by April 30, 1994.

The unapproved vendor technical document has been forwarded to the Vendor Information Control System section and is currently in the approval cycle. There are currently a total of 33 temporary modifications installed at the Cook Nuclear Plant. After a preliminary review, two modifications were identified whose failure could cause a unit trip. Plant Engineering will conduct an independent design verification of these two temp mods. This review will be completed by January 31, 1994.

In the interim, the Plant Engineering Superintendent is taking precautions by closely scrutinizing design complexity of all temp mods that are submitted for Plant Engineering review, and will require independent technical verification if deemed necessary.

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MARTIN,J.B. Document Control Branch (Document Control Desk)

SUBJECT: Submits response to violations noted in Insp Repts 50-315/93-12 & 50-316/93-12.Corrective actions:licensee failed to perform evaluation to determine changes made to feedwater pump speed control sys by temporary mod.

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

INDIANA
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AEP:NRC:1184H
10 CFR 2.201

Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NRC INSPECTION REPORTS NO. 50-315/93012 (DRS)
AND 50-316/93012 (DRS)
REPLY TO NOTICE OF VIOLATION

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

Attn: Mr. J. B. Martin

January 4, 1994

Dear Mr. Martin:

This letter is in response to a letter from G. E. Grant dated November 24, 1993, which forwarded a Notice of Violation associated with a System Based Instrumentation and Control Inspection conducted by Zelig Falevits and others of your office during August 17 through September 28, 1993. The violations are associated with errors in a setpoint calculation for refueling water storage tank level instrumentation and for the lack of an unreviewed safety question determination for Temporary Modification 2-93-015, which installed an I to I converter in the Unit 2 feedwater pump control circuitry.

Our reply to the Notice of Violation is contained in Attachment 1. We were also requested to respond to open and unresolved items from the inspection report. Our response is contained in Attachment 2. In addition, we were requested to address non-cited deficiencies associated with Temporary Modification 2-93-015. Our response to this request is contained in Attachment 3.

This letter is submitted pursuant to 10 CFR 50.54(f) and, as such, an oath statement is attached.

Sincerely,

E.E. Fitzpatrick

E. E. Fitzpatrick
Vice President

dr

100075

Attachments

IE01
1/1

cc: A. A. Blind
G. Charnoff
T. E. Murley - NRC
NFEM Section Chief
NRC Resident Inspector
J. R. Padgett

STATE OF OHIO)
COUNTY OF FRANKLIN)

E. E. Fitzpatrick, being duly sworn, deposes and says that he is the Vice President of licensee Indiana Michigan Power Company, that he has read the forgoing response to the NOTICE OF VIOLATION FOR NRC INSPECTION REPORTS NO. 50-315/93012 (DRS) AND 50-316/93012 (DRS) and knows the contents thereof; and that said contents are true to the best of his knowledge and belief.

E E Fitzpatrick

Subscribed and sworn to before me this 4th

day of January, 19 94.

Rita D. Hill
NOTARY PUBLIC

RITA D. HILL
NOTARY PUBLIC, STATE OF OHIO
MY COMMISSION EXPIRES 6-28-94

ATTACHMENT 1 TO AEP:NRC:1184H

REPLY TO NOTICE OF VIOLATION

NRC Violation I (Severity Level IV)

"10 CFR 50, Appendix B, Criterion III, states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, drawings, procedures, and instructions and that the design control measures shall provide for verifying the adequacy of the design.

Contrary to the above, on September 1, 1993, the team noted that:

a. RWST instrumentation loop setpoint calculation No. 1-2-I9-03, dated August 25, 1993:

   (1) Erroneously derived the setpoint uncertainty value based on the use of Model N-E13 RWST transmitters. However, the installed RWST level transmitters were Model E13DM-HSAH1.

   (2) Provided no justification for using transmitter elevation 599'3" to derive the setpoint uncertainty value in the setpoint calculation.

   (3) Did not consider the error effects of the velocity head of the ECCS pump flow (Safety Injection and Residual Heat Removal pumps) during design basis accidents.

b. Flow diagram OP-1-5144-13, "Containment Spray System Unit #1," incorrectly identified RWST level transmitter ILS-950 as having a minimum level alarm at 638'11". However, the design basis setpoint value was 637' 2"."

Response to Violation I

1. <u>Admission or Denial of the Alleged Violation</u>

   Indiana Michigan Power admits to the violation as cited in the NRC Notice of Violation.

2. <u>Reasons for the Violation</u>

   The examples of the violation will be addressed individually.

   1.a.(1) The cause of the violation is attributed to lack of attention to detail in cross-checking the specific details in the documentation. The instrument data sheet used was from a slightly different model (N-E13 versus the correct model E13 of the same vendor). Both types of instruments are used in the plant and have nearly identical performance characteristics. The correct model data sheet was compared to the incorrect model data sheet and minor differences in some of the uncertainty terms were found which required the

calculation to be revised. However, no significant differences were found that changed the end results of the calculation adversely. Therefore this error did not have an adverse effect on safety.

1.a.(2) As noted in Revisions 2 and 4 of the calculation, the elevations used in the calculation were based on field walkdown measurements, which were taken in 1979. During the inspection, the elevations were remeasured but did not exactly match the previous field walkdown measurements. Because of the time that has elapsed, it was not possible to positively identify the reason for the discrepancy. It is noted, however, that the differences between the as-built measurements taken during the inspection and the measurements used in the calculation differed only slightly. The differences in elevation between the as-built heights and the heights used in the calculation varied between 1 inch and 2.5 inches, compared to an instrument span of 363 inches. The worst case error in the non-conservative direction (as-built lower than calculation) was 2.5 inches, which equates to an error of approximately 0.7%. In other words, the RWST indication and alarm will occur 0.7% lower than actual level. However, since there is considerable margin built into the alarm setpoints (approximately 21% for the low alarm and approximately 7% for the low-low alarm), this measurement error had no adverse effect on safety.

1.a.(3) The alarms and trips associated with the RWST level instrumentation include both high and low level type functions. There are two high level functions. The High Level Alarm is used to ensure the operators do not inadvertently overflow the RWST when filling, and a Minimum Level alarm is used to alert the operators that Tech Spec required minimum RWST volume requirements are being encroached. There are also two low level functions. The Low Level alarm is used to alert the operator that RWST inventories are approaching a low level at which the operator should begin to transfer to ECCS recirculation mode, and a Low-Low Level alarm and RHR pump trip is used to alert the operator that RWST inventory is depleted and to protect the RHR pump from damage due to low NPSH.

Because the RWST level transmitter is tapped off the ECCS suction line at the bottom of RWST tank, velocity head effects can be induced when the ECCS pumps are running. The high level functions are not affected by this because these functions are only used when the ECCS pumps are not running. The low level functions are affected because the ECCS pumps are running when they are required to function. However,

computation of the velocity head effect shows it will only affect the low level functions in a conservative manner. The velocity head effect induces a negative bias that results in a level indication that is lower than actual level. The effect on the low level alarms and RHR pump trip functions is that they will occur sooner and therefore does not jeopardize the RHR pump or plant safety.

Velocity head effect was not addressed due to unawareness of its influence on this application. Criteria for when this effect is to be considered were not included in the procedure used for the preparation of this calculation. It should be noted that most tank level instruments are tapped on the side of the tank and are therefore not affected by the velocity head effects of the tank suction line.

1.b. Setpoint information for the Cook Nuclear Plant is controlled through the Plant Setpoint Document, rather than through the flow prints. Because of this, there was no systematic process to have setpoints placed on flow prints, or to update the flow prints if the setpoint changed. The setpoints displayed on the drawings are for reference and are used for understanding the drawing only.

3. <u>Corrective Actions Taken and Results Achieved</u>

The examples of the violation will be addressed individually.

1.a.(1) The correct model data sheet has been compared with the incorrect model instrument data sheet and no significant differences were found. The calculation will be revised to incorporate the correct instrument data sheet information.

1.a.(2) The calculation has been revised to reflect the correct as-built elevations for the RWST level transmitters.

1.a.(3) The calculation has been revised to address the velocity head effects. Review of other tank level applications found the CST tank level to have a similar velocity head effect which was not addressed. The CST calculation will also be corrected.

1.b. Drawing OP-1-5144 and OP-2-5144 were revised to reflect the correct setpoints.

4. Actions Taken to Avoid Further Violations

As a general comment, it is noted that numerous instrument setpoint calculations are being updated as part of the Reactor Protection and Control Systems Upgrade Project which will be implemented during the 1994 refueling outages. The specific examples of the violation are addressed individually, below.

1.a.(1) Training sessions were held for Corporate I&C engineers which emphasize that self-checking and engineering reviews and verification are expected to be such that errors in instrument model numbers and similar documentation data is discovered and corrected prior to document issue.

1.a.(2) A sampling of safety-related instruments will be checked to verify correct as-built elevations are incorporated into setpoint calculations. The sampling will be completed by May 31, 1994. Further preventive measures will be established depending on the results of the sampling.

1.a.(3) The engineering guide governing setpoint calculations was revised on November 15, 1993, to require that process measurement effects, such as velocity head effects, are considered, as necessary, as part of the calculation preparation.

1.b. Since the intent of setpoint information on flow prints is only to help in the understanding of the drawing, a note will be added to all affected flow prints which states:

"Caution! The setpoints indicated are provided only to assist in understanding the drawing. Refer to appropriate setpoint control document for actual device setpoints." The notes will be added by July 6, 1994.

5. Date When Full Compliance will be Achieved

The examples of the violation will be addressed individually.

1.a.(1) Full compliance will be achieved by January 10, 1994, when the calculation is revised to reflect the correct model performance characteristics.

1.a.(2) Full compliance was achieved on November 5, 1993, when the calculation was revised to incorporate the as-built transmitter elevations.

1.a.(3) Full compliance was achieved on November 5, 1993, when the calculation was revised to include consideration of velocity head effects. The calculation for the CST will be revised by January 10, 1994, to correct the similar deficiency identified during the review of the RWST calculation.

1.b. Full compliance was achieved on September 15, 1993, when the affected Unit 1 and 2 OP drawings were revised.

NRC Violation II (Severity Level IV)

"10 CFR 50.59 states licensees may make changes to the facility as described in the safety analysis report without prior Commission approval unless the change involves an unreviewed safety question. A written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question is required.

Section 10.5.1.1 of the UFSAR states that the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators. In addition, Section 14.1.9 analyzed a loss of normal feedwater from pump failures which could result in a reduction of the secondary system to remove heat generated in the reactor core.

Contrary to the above, on April 7, 1993, the licensee failed to perform an evaluation to determine that changes made to the feedwater pump speed control system by temporary modification 2-93-015 did not involve an unreviewed safety question."

Response to Violation II

1. <u>Admission or Denial of the Alleged Violation</u>

   Indiana Michigan Power denies the violation as cited in the NRC Notice of Violation.

2. <u>Reasons for Denial of the Violation</u>

   At the Cook Nuclear Plant, temporary modifications undergo a screening to determine if an unreviewed safety question determination is required to be performed pursuant to 10 CFR 50.59. The process we use is based on the guidance of NSAC 125 (June 1989), entitled "Guidelines for 10 CFR 50.59 Safety Evaluations." This document was prepared jointly by the Nuclear Management and Resources Council (NUMARC) and the Nuclear Safety Analysis Center of the Electric Power Research Institute (EPRI).

   The inspection report (page 14) states:

   > "By failing to recognize that the speed control system was described in the UFSAR, the licensee concluded that 10 CFR 50.59 was not applicable, therefore, no safety evaluation was performed. The licensee's failure to perform a safety evaluation is considered to be a violation of 10 CFR 50.59."

We disagree with the statement that the speed control system is described in the UFSAR. The UFSAR (Section 10.5.1.1) specifically states that "the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators." There is no description of the circuitry provided in this statement.

NSAC 125 recognizes that changes made to the facility may implicitly impact the UFSAR. For these cases, NSAC 125 states that:

> "If the SSC (structure, system, or component) is part of a larger SSC described in the SAR and if the change affects the design, function, or method of performing the function of the larger SSC AS DESCRIBED IN THE SAR (emphasis added) then a safety evaluation is required."

As discussed in the UFSAR, the variable speed turbine driven main feedwater pumps are designed to provide the required feedwater flow to the steam generators. The temporary modification only added a device to isolate noise in the speed control circuitry. Neither the function of the main feedwater pumps as described in the UFSAR (to provide feedwater flow to the steam generators) nor the method of performing the function as described in the UFSAR (with variable speed pumps) were impacted by the temporary modification. Additionally, since there was no description of the feedwater control system in the UFSAR, neither the design of the control system nor the design of the feedwater system as a whole as described in the UFSAR were impacted by the change. Based on these considerations, we conclude that no unreviewed safety question determination was required pursuant to 10 CFR 50.59.

As described in NSAC 125,

> "The purpose of 10 CFR 50.59 is to preserve the original licensing basis in the information submitted to the NRC as part of the application for an operating license and in the final safety evaluation report (SER) issued by the NRC staff. The NRC relies on this information to conclude that an operating license can be issued without undue risk to the health and safety of the public. This regulation allows the licensee to make changes without prior NRC approval while maintaining the licensing basis. It defines conditions that must be met in determining if prior regulatory review is needed."

The level of detail included in the UFSAR regarding the feedwater system is relatively small. This is commensurate with the fact that the system is non-safety related. The NRC staff review of the system during the original licensing of the plant would be expected to be of a different level of detail than for those systems which are safety related. Since the staff review did not rely on a detailed description of the speed

control system in order to conclude that the feedwater system was adequate from a safety perspective, it is not reasonable to conclude that 10 CFR 50.59 would be applicable to changes to the circuitry. It is noted that Section 14.1.9 of the UFSAR, entitled "Feedwater System Malfunctions," analyzes a complete loss of feedwater (due to no specific reason) and concludes that:

> "a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves."

It is also noted (as acknowledged in the inspection report, page 14) that failure of the I to I converter would have the same effect as the failure of the hand/auto station already in the circuit. In other words, the change did not introduce a new failure mode into the system.

ATTACHMENT 2 TO AEP:NRC:1184H

RESPONSE TO OPEN AND UNRESOLVED ITEMS

The cover letter for Inspection Report 50-315/316 93-012 (DRS) requested we provide a written response to open and unresolved items in the inspection report. There is only one item in this category, Unresolved Item 315/316 93012-04(DRS). This unresolved item involves a calculation performed to verify that an instrument sensing line associated with the Unit 2 auxiliary feedwater system was adequately supported. The inspection report states that:

> "The licensee was in the process of performing a calculation to determine whether the sensing line installation was adequate. Pending review of the calculation, this item is considered unresolved."

The subject calculation was completed following the inspection, and, at the inspector's request the information was mailed in an overnight package to NRC Region III on October 20, 1993. The information, documented in a memo from S. J. Jarrett/H. P. Damasco to M. S. Ackerman dated October 19, 1993, follows.

AEP
AMERICAN ELECTRIC POWER

Date October 19, 1993

Subject Cook Nuclear Plant
NRC SBICI Inspection Assistance
NEDS Review of As-Found Conditions

From S. J. Jarrett/H. P. Damasco

To M. S. Ackerman

As per your request, the Nuclear Design - Structural & Analytical Section (NEDS) has reviewed the as-found condition detected by an NRC inspector during a NRC SBICI Inspection. The 3/8"ϕ, 1/2"ϕ, and 3/4"ϕ downstream low pressure instrument lines associated with 2-FFS-257, and branch lines off the feed water header line 2-FW-48 were believed to have piping/tubing spans that exceeded Alternate Analysis Criteria.

Based on the above findings, Nuclear Engineering Site Design Section (NESD) performed a detailed walkdown of the piping system (see Attachment A). In order to satisfy the seismic overlap criteria, NESD included additional piping and supports beyond the area addressed by the NRC inspector. In so doing, they found that two support components in the overlap region of the continuation line were missing.

Nuclear Design - Mechanical Section (NEDM), performed a preliminary assessment of the piping system as found by the NRC inspector and concluded in their E-Mail, dated September 17, 1993 to Stan Farlow (see Attachment B), that the minor deviations in some support span lengths have an insignificant effect on the operability and/or the design basis of the instrument line.

NEDS performed the as-found and the as-designed DBE seismic evaluations on the piping system, in Calculation No. DC-D-02-MSC-36, by using walkdown information (see Attachment A), and Ebasco/P-Delta (E/PD) STRUDL integrated computer program. The piping stresses and the valve accelerations in these analyses were found to be well within the design basis criteria allowable limits. Attachments C through F highlight the very small stress interaction ratios, and vertical and horizontal valve acceleration ratios resulting from the E/PD STRUDL analyses. These analyses were performed to confirm the NEDM preliminary engineering review.

Although the piping system in the as-found condition meets the design basis criteria limits, the two missing components are being installed on the piping system to reflect similarity of pipe supports on identical piping systems in the vicinity, and further to conform with good engineering practice.

It is NEDS understanding that Job Order No. C19393 has been initiated by Plant Maintenance to install the two missing components. If you have any further questions please contact these writers at extension 3157.

S. J. Jarrett/H. P. Damasco

Approved by

FOR N. Ruccia, Manager
Nuclear Design - Structural & Analytical Section

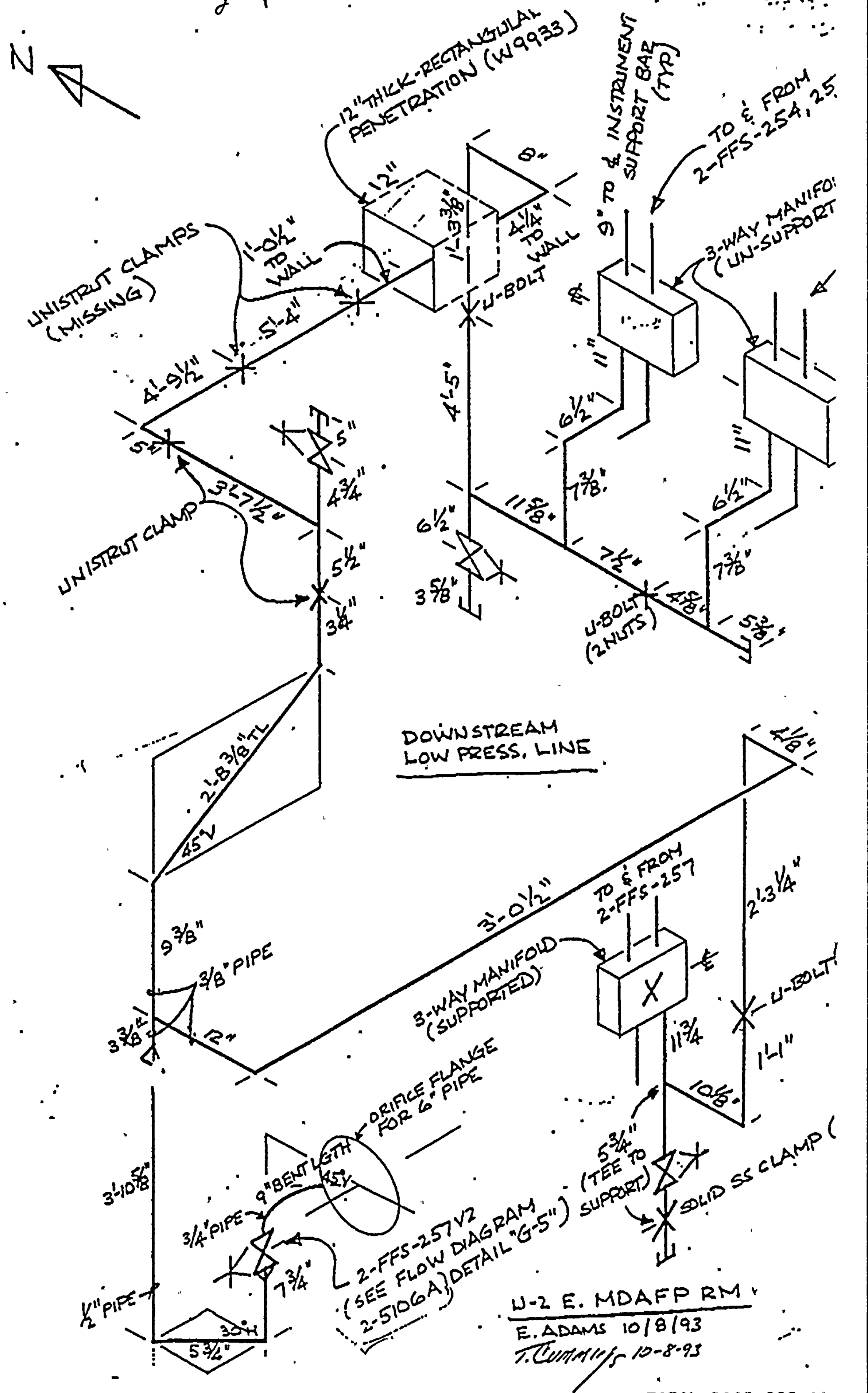
/sjj

xc: R. C. Armstrong/S. P. Hodge - w/o attachment
A. K. Dey - w/o attachment
S. K. Farlow - w/o attachment
R. C. Carruth - w/o attachment
T. H. Cummings - w/o attachment

Attachment A

Copy of Fieldwalk from NESD

2-FW-48


12" THICK-RECTANGULAR PENETRATION (W9933)
UNISTRUT CLAMPS (MISSING)
1'-0½" TO WALL
4¼" TO WALL
9" TO ℄ INSTRUMENT SUPPORT BAR (TYP)
TO ℄ FROM 2-FFS-254, 25
3-WAY MANIFOLD (UN-SUPPORT
U-BOLT
4'-9½"
5'-4"
4'-5"
UNISTRUT CLAMP
3'-7½"
U-BOLT (2 NUTS)
DOWNSTREAM LOW PRESS. LINE
2'-8⅜" TL
45°
9⅜"
⅜" PIPE
3'-0½"
TO ℄ FROM 2-FFS-257
3-WAY MANIFOLD (SUPPORTED)
2'-3¼"
U-BOLT
ORIFICE FLANGE FOR 6" PIPE
9" BENT LGTH
3/4" PIPE
½" PIPE
3'-10⅝"
2-FFS-257V2 (SEE FLOW DIAGRAM 2-5106A) DETAIL "G-5")
5¾" (TEE TO SUPPORT)
SOLID SS CLAMP (
U-2 E. MDAFP RM
E. ADAMS 10/8/93
T. Cummins 10-8-93

Attachment B

Copy of E-Mail to Stan Farlow (dated 9/17/93) from Amiya Dey

To: STAN K. FARLOW@NED@AEPSC
ROBERT C. CARRUTH@NED@AEPSC
Thomas H Cummings@SDO@COOK,STEVE P. HODGE@NED@AEPSC
From: AMIYA K. DEY@NDG@AEPSC
Subject: NRC Inquiry Relative To 2-FFS-257
Date: Friday, September 17, 1993 13:44:19 EDT
Attach:
Certify: N
Forwarded by:

Instrument Line 2-FFS-257

I understand that an NRC inspector has noted the piping line associated with the Instrument #2-FFS-257 to be under supported as compared to MDS-601 design standard.
Our findigs are given below:
The Standard MDS-601 was issued in Feb.1988 and I believe this instrument line was installed much earlier.This system and other instrument piping/tubing systems were installed per EDS Corp. developed guideline titled"Field Fabricated Support Guide For Small Diameter Pipes and Copper Tubing".This guideline was included as a section in a larger guideline known as the Alternate Analysis Criteria(AAC).
The AAC provides cookbook type directions in supporting piping/tubing systems and is very conservative.Our experience, with piping/tubing systems analysed and supported per AAC,indicates that:

*these systems generally are well designed and meet applicable code allowables easily when computer analysis is performed to verify system adequacy where limitations of the AAC have been marginally exceeded.

The instrument line being reviewed is 1/2 in. sch.80 pipe upto a TEE connection and then reduces to 3/8 in.sch. pipe sections.Our walkdown review of this systems indicates minor deviations in some span lenghths which we believe have no significance on the operability and/or the design basis of this insturment line.
If you need additional information,please contact me at 3860.
Thanks,
☺ Amiya ☺

Attachment C

Methodology

Method used to analyze the piping system in Calculation No. DC-D-02-MSC-36

AEPSC
DONALD C. COOK NUCLEAR PLANT , UNIT 2

6.0 METHODOLOGY

The piping system has been rigorously analyzed using walkdown information and Ebasco/P-Delta (E/PD) STRUDL integrated computer program. Additional hand calculations, where necessary, have been performed to develop problem input data and/or to justify acceptance of the piping system.
The information relevant to the final output is given below:

COMPUTER PROGRAM & VERSION:

E/PD STRUDL Version 0193 VAX VMS 5.5

JOB No.	RUN DATE & TIME	COMMENTS
2448	10-18-93 08:21	AF run - see comm. below
2449	10-18-93 08:23	DSN run - see comm. below

ADDITIONAL COMMENTS:

Job no. 2448 is the run simulating the as-found condition with Design Basis Earthquake response spectra.

Job no. 2449 is the as-design run that includes the two missing pipe clamps, and using the Design Basis Earthquake response spectra.

REV.	BY	DATE	CHK.	DATE
0	[signature]	10/18/93	[signature]	10/18/93

NUCLEAR DESIGN GROUP
CALCULATION No. DC-D-02-MSC-36

Attachment D

Problem Description

AEPSC
DONALD C. COOK NUCLEAR PLANT, UNIT 2

7.0 PROBLEM DESCRIPTION

GENERAL DESCRIPTION:

Analysis on the as-found 3/8", 1/2", 3/4" downstream low pressure instrument header and branch lines off feedwater line 2-FW-48 to address overspan concerns raised by an NRC inspector.

NUMBER OF SUPPORTS:

8 (including the two missing unistrut clamps)

PRIMARY BUILDING:

Turbine - Area 2T4

ELEVATION OF MAIN PIPING:

593'-0"

MAJOR EQUIPMENT:

None

0	[illegible]	10/18/93	[illegible]	10/18/93
REV.	BY	DATE	CHK.	DATE

NUCLEAR DESIGN GROUP

CALCULATION No. DC-D-02-MSC-36

Attachment E

This attachment shows stress ratios, and valve acceleration ratios, from the <u>as-found analysis</u> which excludes the two missing support components, that are well within the design basis criteria allowable limits.

Sheet 1	Stress Interaction Ratio
Sheets 2&3	Horizontal Valve Acceleration Ratios
Sheets 4&5	Vertical Valve Acceleration Ratios

//-PIPING-/-TBLNAM-/--CODE--/--------/--NODE--/EQUATION/				------APPLIED FORCES-----/		--------STRESS DATA--------		//--LOAD--/	-RESULT-//
MEMBER	SECNAM	VERSION		FX/MX	FY/MY	FZ/MZ	PRES/NONP	TOT/ALW	RATIO	
			31.1-4	0.0	0.0	0.0	826.22	7335.86		PASS
				0.0	0.0	0.0	6509.64	37500.00	0.20	

SUMMARY OF THE PIPING MEMBER CHECKS BY EQUATION

EQUATION	MAXIMUM IR	MEMBER	LOAD
31.1-1	0.4748	98	13
31.1-2NU	0.5887	44	1000
31.1-2E	0.5290	44	2000
31.1-3	0.0979	30	14
31.1-4	0.0979	98	

INTERACTION RATIOS

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 54 PIPE MEMBERS, THAT ARE CHECKED, PASSED CODE CHECKS.

```
$CHECK PIPE BREAK LOC ALL
$CHECK PIPE CRACK LOC ALL
$CHECK PIPE BRANCH DISPL ALL
 PRINT PIPE ADD STRESS INT FACT
```

Attachment E
Sheet 1 of 5

****
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE 2, RESULTS *
****

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

VALVE NODE	TBLNAM SECNAM	TITLE	CRITICAL LOAD	MEMBER	APPLIED DATA AX/MX	APPLIED DATA AY/MY	APPLIED DATA AZ/MZ	ALLOWABLE DATA ACC	ALLOWABLE DATA MOM	RATIO	RESULT EQUATION
308		GATE1	123	308L	0.00 0.00	27.39 0.00	0.00 0.00	772.80	1000000.06	0.04	PASS ONE
			123	308M	0.00 0.00	27.39 0.00	0.00 0.00	772.80	1000000.06	0.04	PASS ONE
			456	308L	0.01 0.00	54.24 0.00	0.01 0.00	772.80	1000000.06	0.07	PASS ONE
			456	308M	0.01 0.00	54.24 0.00	0.01 0.00	772.80	1000000.06	0.07	PASS ONE
508		GATE3	123	508L	0.00 0.00	27.39 0.00	0.03 0.00	772.80	1000000.06	0.04	PASS ONE
			123	508M	0.00 0.00	27.39 0.00	0.03 0.00	772.80	1000000.06	0.04	PASS ONE
			456	508L	0.01 0.00	54.24 0.01	0.05 0.00	772.80	1000000.06	0.07	PASS ONE
			456	508M	0.01 0.00	54.24 0.01	0.05 0.00	772.80	1000000.06	0.07	PASS ONE
608		GATE2	123	608L	0.00 0.00	49.27 0.00	0.00 0.00	772.80	1000000.06	0.06	PASS ONE

Attachment E
Sheet 2 of 5

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA AX/MX	AY/MY	AZ/MZ ----------/	----ALLOWABLE DATA ACC	MOM ---/	--RATIO-/	-RESULT-// EQUATION
			123	608M	0.00	49.27	0.00	772.80	1000000.06	0.06	PASS
					0.00	0.00	0.00				ONE
			456	608L	0.01	97.46	0.01	772.80	1000000.06	0.13	PASS
					0.00	0.00	0.00				ONE
			456	608M	0.01	97.46	0.01	772.80	1000000.06	0.13	PASS
					0.00	0.00	0.00				ONE
438		Y075T38	123	438L	0.00	27.32	0.00	772.80	1000000.06	0.04	PASS
					0.01	0.02	0.02				ONE
			123	438M	0.00	27.32	0.00	772.80	1000000.06	0.04	PASS
					0.01	0.02	0.02				ONE
			456	438L	0.01	54.06	0.01	772.80	1000000.06	0.07	PASS
					0.02	0.04	0.04				ONE
			456	438M	0.01	54.06	0.01	772.80	1000000.06	0.07	PASS
					0.02	0.04	0.04				ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
$
$CHECK VALVE HORIZONT. ACCEL. FOR 3G
$
PIPE VALVE CHECK PARA
AXPFF 1.0000 LOADS ALL
AYPFF 0.0001 LOADS ALL
AZPFF 1.0000 LOADS ALL
AAF   1.5000 JOI   ALL
TRACE 2
END
LOAD LIST  1000  2000
CHECK PIPE VALVE INT EQU JOI -
   308        508        608        438 -
 NOCH TWO
```

Attachment E
Sheet 3 of 5

****
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE 2, RESULTS *
****

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA---------- AX/MX	AY/MY	AZ/MZ/	----ALLOWABLE DATA----/ ACC	MOM	-RATIO-/	-RESULT-// EQUATION
308		GATE1	1000	308L	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
			1000	308M	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
			2000	308L	81.58 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
			2000	308M	81.58 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
508		GATE3	1000	508L	40.48 0.00	0.00 0.00	259.97 0.00	1159.20	1000000.06	0.23	PASS ONE
			1000	508M	40.48 0.00	0.00 0.00	259.97 0.00	1159.20	1000000.06	0.23	PASS ONE
			2000	508L	80.16 0.00	0.01 0.01	524.42 0.00	1159.20	1000000.06	0.46	PASS ONE
			2000	508M	80.16 0.00	0.01 0.01	524.42 0.00	1159.20	1000000.06	0.46	PASS ONE
608		GATE2	1000	608L	40.89 0.00	0.00 0.00	46.15 0.00	1159.20	1000000.06	0.05	PASS ONE

//-VALVE--/-TBLNAM-/------TITLE------/CRITICAL/-MEMBER-/				----------APPLIED DATA----------/			----ALLOWABLE DATA----/		--RATIO-/	-RESULT-//
NODE	SECNAM	TITLE	LOAD	MEMBER	AX/HX	AY/HY	AZ/HZ	ACC	MOM	RATIO	EQUATION
			1000	608H	40.89 0.00	0.00 0.00	46.15 0.00	1159.20	1000000.06	0.05	PASS ONE
			2000	608L	80.98 0.00	0.01 0.00	90.76 0.00	1159.20	1000000.06	0.10	PASS ONE
			2000	608H	80.98 0.00	0.01 0.00	90.76 0.00	1159.20	1000000.06	0.10	PASS ONE
438		Y075T38	1000	438L	39.75 0.04	0.00 0.12	45.44 0.07	1159.20	1000000.06	0.05	PASS ONE
			1000	438H	39.75 0.04	0.00 0.12	45.44 0.07	1159.20	1000000.06	0.05	PASS ONE
			2000	438L	78.19 0.05	0.01 0.14	89.34 0.09	1159.20	1000000.06	0.10	PASS ONE
			2000	438H	78.19 0.05	0.01 0.14	89.34 0.09	1159.20	1000000.06	0.10	PASS ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
PIPING LOAD TYPES ALEVEL     11   701 -
BLEVEL   1000 TO   1001 CLEVEL   2000 TO   2001 DEAD 11 -
THERMAL      1
 SECTION FR  NS 2  0.0  1.0
 OUTPUT DECIMAL 5
 OUTPUT BY MEMBERS
 LOAD LIST     11     1 701 1000 TO 1001 2000 TO 2001
$
UNIT INCH POUND RAD
LIST DISPLACEMENTS '&PIPEJOI'
```

Attachment E
Sheet 5 of 5

Attachment F

This attachment shows stress ratios, and valve acceleration ratios, from the as-design analysis which includes the two missing support components, that are well within the design basis criteria allowable limits.

Sheet 1	Stress Interaction Ratio
Sheets 2&3	Horizontal Valve Acceleration Ratios
Sheets 4&5	Vertical Valve Acceleration Ratios

```
//-PIPING-/-TBLNAM-/--CODE--/--------/--NODE--/EQUATION/------APPLIED FORCES-----/--------STRESS DATA--------//--LOAD--/-RESULT-//
   MEMBER   SECNAM  VERSION                                FX/MX    FY/MY    FZ/MZ     PRES/NONP   TOT/ALW           RATIO

                                               31.1-4      0.0      0.0      0.0       826.22     7368.35                  PASS
                                                           0.0      0.0      0.0      6542.13    37500.00           0.20
```

SUMMARY OF THE PIPING MEMBER CHECKS BY EQUATION

EQUATION	MAXIMUM IR	MEMBER	LOAD
31.1-1	0.4766	98	13
31.1-2NU	0.4753	98	1000
31.1-2E	0.4770	44	2000
31.1-3	0.0973	30	14
31.1-4	0.0973	98	

INTERACTION RATIOS

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 54 PIPE MEMBERS, THAT ARE CHECKED, PASSED CODE CHECKS.

```
$CHECK PIPE BREAK LOC ALL
$CHECK PIPE CRACK LOC ALL
$CHECK PIPE BRANCH DISPL ALL
 PRINT PIPE ADD STRESS INT FACT
```

Attachment F
Sheet 1 of 5

```
*********************************************************
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE  2, RESULTS *
*********************************************************
```

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	----------APPLIED DATA---------- AX/MX	AY/MY	AZ/MZ	/----ALLOWABLE DATA----/ ACC	MOM	--RATIO-/	-RESULT-// EQUATION
308		GATE1	123	308L	0.00 0.00	27.39 0.00	0.00 0.00	772.80	1000000.06	0.04	PASS ONE
			123	308M	0.00 0.00	27.39 0.00	0.00 0.00	772.80	1000000.06	0.04	PASS ONE
			456	308L	0.01 0.00	54.24 0.00	0.01 0.00	772.80	1000000.06	0.07	PASS ONE
			456	308M	0.01 0.00	54.24 0.00	0.01 0.00	772.80	1000000.06	0.07	PASS ONE
508		GATE3	123	508L	0.00 0.00	27.39 0.00	0.03 0.00	772.80	1000000.06	0.04	PASS ONE
			123	508M	0.00 0.00	27.39 0.00	0.03 0.00	772.80	1000000.06	0.04	PASS ONE
			456	508L	0.01 0.00	54.24 0.01	0.06 0.00	772.80	1000000.06	0.07	PASS ONE
			456	508M	0.01 0.00	54.24 0.01	0.06 0.00	772.80	1000000.06	0.07	PASS ONE
608		GATE2	123	608L	0.00 0.00	51.23 0.00	0.00 0.00	772.80	1000000.06	0.07	PASS ONE

Attachment F
Sheet 2 of 5

//-VALVE--/ NODE	-TBLNAM-/ SECNAM	------TITLE------/	CRITICAL/ LOAD	-MEMBER-/	AX/MX	AY/MY	AZ/MZ	ACC	MOM	--RATIO-/	-RESULT-// EQUATION
			123	608H	0.00 0.00	51.23 0.00	0.00 0.00	772.80	1000000.06	0.07	PASS ONE
			456	608L	0.01 0.00	101.30 0.00	0.01 0.00	772.80	1000000.06	0.13	PASS ONE
			456	608H	0.01 0.00	101.30 0.00	0.01 0.00	772.80	1000000.06	0.13	PASS ONE
438		Y075T38	123	438L	0.00 0.01	27.64 0.02	0.00 0.02	772.80	1000000.06	0.04	PASS ONE
			123	438H	0.00 0.01	27.64 0.02	0.00 0.02	772.80	1000000.06	0.04	PASS ONE
			456	438L	0.01 0.02	54.70 0.04	0.01 0.04	772.80	1000000.06	0.07	PASS ONE
			456	438H	0.01 0.02	54.70 0.04	0.01 0.04	772.80	1000000.06	0.07	PASS ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
$
$CHECK VALVE HORIZONT. ACCEL. FOR 3G
$
PIPE VALVE CHECK PARA
AXPFF 1.0000 LOADS ALL
AYPFF 0.0001 LOADS ALL
AZPFF 1.0000 LOADS ALL
AAF   1.5000 JOI   ALL
TRACE 2
END
LOAD LIST  1000  2000
CHECK PIPE VALVE INT EQU JOI -
   308        508        608        438 -
 NOCH TWO
```

Attachment F
Sheet 3 of 5

****
* STRUDL PIPING VALVE MEMBER CHECK ,TRACE 2, RESULTS *
****

JOB ID - FFS257 JOB TITLE - CALCULATION No. DC-D-02-MSC-36

ACTIVE UNITS -	LENGTH	WEIGHT	ANGLE	TEMPERATURE	TIME	MASS
	INCH	KIPF	RAD	FAH	SEC	LBM

//-VALVE--/-TBLNAM-/	------TITLE------/	CRITICAL/	-MEMBER-/	----------APPLIED DATA----------		/	----ALLOWABLE DATA----/		--RATIO-/	-RESULT-//
NODE	SECNAM		LOAD		AX/MX	AY/MY	AZ/MZ	ACC	MOM		EQUATION
308		GATE1	1000	308L	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
			1000	308M	41.19 0.00	0.00 0.01	41.10 0.00	1159.20	1000000.06	0.05	PASS ONE
			2000	308L	81.56 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
			2000	308M	81.56 0.00	0.01 0.01	81.39 0.00	1159.20	1000000.06	0.10	PASS ONE
508		GATE3	1000	508L	40.63 0.00	0.00 0.00	308.47 0.00	1159.20	1000000.06	0.27	PASS ONE
			1000	508M	40.63 0.00	0.00 0.00	308.47 0.00	1159.20	1000000.06	0.27	PASS ONE
			2000	508L	80.47 0.00	0.01 0.01	620.58 0.00	1159.20	1000000.06	0.54	PASS ONE
			2000	508M	80.47 0.00	0.01 0.01	620.58 0.00	1159.20	1000000.06	0.54	PASS ONE
608		GATE2	1000	608L	40.76 0.00	0.01 0.00	48.66 0.00	1159.20	1000000.06	0.05	PASS ONE

VALVE NODE	TBLNAM SECNAM	TITLE	CRITICAL LOAD	MEMBER	APPLIED DATA AX/HX	AY/HY	AZ/HZ	ALLOWABLE DATA ACC	MOM	RATIO	RESULT EQUATION
			1000	608H	40.76 0.00	0.01 0.00	48.66 0.00	1159.20	1000000.06	0.05	PASS ONE
			2000	608L	80.72 0.00	0.01 0.00	95.58 0.00	1159.20	1000000.06	0.11	PASS ONE
			2000	608H	80.72 0.00	0.01 0.00	95.58 0.00	1159.20	1000000.06	0.11	PASS ONE
438	Y075T38		1000	438L	41.17 0.03	0.00 0.11	45.88 0.07	1159.20	1000000.06	0.05	PASS ONE
			1000	438H	41.17 0.03	0.00 0.11	45.88 0.07	1159.20	1000000.06	0.05	PASS ONE
			2000	438L	80.98 0.05	0.01 0.13	90.20 0.09	1159.20	1000000.06	0.10	PASS ONE
			2000	438H	80.98 0.05	0.01 0.13	90.20 0.09	1159.20	1000000.06	0.10	PASS ONE

********** FOLLOWING IS A SUMMARY OF THE CODE CHECKS PERFORMED ABOVE **********

ALL 4 PIPING VALVE JOINTS , THAT ARE CHECKED, PASSED CODE CHECKS.

```
PIPING LOAD TYPES ALEVEL     11   701 -
BLEVEL   1000 TO   1001 CLEVEL   2000 TO   2001 DEAD 11 -
THERMAL      1
 SECTION FR  NS 2  0.0  1.0
 OUTPUT DECIMAL 5
 OUTPUT BY MEMBERS
 LOAD LIST     11     1 701 1000 TO 1001 2000 TO 2001
$
UNIT INCH POUND RAD
LIST DISPLACEMENTS '&PIPEJOI'
```

Attachment F
Sheet 5 of 5

ATTACHMENT 3 TO AEP:NRC:1184H

INFORMATION REGARDING TEMPORARY MODIFICATION

2-93-015

As requested in the cover letter for inspection report 50-315/316 93012 (DRS), this attachment provides the results of our investigation into the issues associated with Temporary Modification (TM) 2-93-015. This TM installed current-to-current (I/I) converter modules in the control circuitry for the Unit 2 East and West Main Feedwater Pumps. The purpose of the temporary modification was to isolate the control circuitry from the presence of signal grounds in the field wiring that were interfering with proper operation of the D/P slave controller. The modification was processed as a temporary modification because the source of the grounding problem could not be resolved with the unit on line. In addition, the control room portion of the feedwater control system circuitry has been scheduled for replacement in 1994 as part of the reactor protection system upgrade project.

Two design errors occurred that were found during the post-installation checkout process. Specifically, the design as released by Plant Engineering had the I/I wired in backwards (i.e., input and output reversed), and a necessary 100 ohm input resistor omitted. These errors were discovered during post-installation circuit checks, prior to post modification testing (PMT) and prior to placing the affected circuitry back into operation. Since the errors were identified and corrected prior to PMT, no condition report was required. Although not required, a condition report was subsequently generated to document our investigation.

The TM did not involve safety related equipment, and therefore the quality assurance requirements of 10 CFR 50 Appendix B and the requirements of ANSI-N45.2.11 do not specifically apply. However, since the temp mod process used at the plant is common to both safety-related and non-safety related changes, our review of this event was conducted to consider its implications for safety-related temp mods.

The following conclusions were drawn from our investigation of this event:

1. The design error occurred as the result of attempting to copy the design details of an existing I/I circuit from another approved plant drawing to the drawing markup which was used to serve as the installation drawing for the temp mod. While possible to correctly identify the input and output leads on this drawing, the input/output lead designations (W,X & D,F) were inadvertently transposed during their transfer to the mark-up. The 100 ohm input resistor was overlooked and was not transferred to the mark-up.

2. A vendor document detailing the design and installation requirements for the I/I was available, but not as an approved plant document. The engineer chose not to use it, because he felt confident that he was adequately familiar with the hardware and that reference to the vendor information would be unnecessary. Use of the document would likely have allowed the engineer to detect the transposition error, as well as the omission of the input resistor.

3. The engineer assigned to develop and implement the temp mod request was adequately qualified to perform this task.

4. The supervisor who reviewed the marked-up drawings was also adequately qualified to perform this function. However, the supervisor became involved with the development of the drawing markups. As a result, his review was not independent, which may have compromised his ability to detect the design errors.

5. The temp mod procedure does not clearly convey design review requirements. The Engineering Supervisor initialed the drawings in the temp mod package indicating his concurrence. However, this review is not a step required by the temp mod procedure.

6. The Plant Engineering review required by the temp mod procedure calls for "...a review of the request to ensure that it is needed, correct, practical, and that it accomplishes the intended purpose." It does not require a technical verification of the design change documents (i.e. drawing mark-up in this case) that are used to implement the request.

This event has identified a weakness in the temp mod procedure design review requirements. In order to prevent recurrence of an incident of this nature, the TM procedure, PMP.5040.MOD.001, will be revised to strengthen the requirements for verifying the technical accuracy of the design. The revision will clearly identify expectations and responsibilities for the review. These revisions will be completed by April 30, 1994.

The unapproved vendor technical document has been forwarded to the Vendor Information Control System section and is currently in the approval cycle. There are currently a total of 33 temporary modifications installed at the Cook Nuclear Plant. After a preliminary review, two modifications were identified whose failure could cause a unit trip. Plant Engineering will conduct an independent design verification of these two temp mods. This review will be completed by January 31, 1994.

In the interim, the Plant Engineering Superintendent is taking precautions by closely scrutinizing design complexity of all temp mods that are submitted for Plant Engineering review, and will require independent technical verification if deemed necessary.