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July 29, 1994

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317; License No. DPR 53
Licensee Event Report 94-004, Revision 01
Excessive Corrosion of Incore Instrumentation Flange Components

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

Charles H. Cruse

CHC/MDM/bjd

Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
M. K. Boyle, NRC
D. G. McDonald, Jr., NRC
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and Program Control

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NRC FORM 366 (5-92) U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 1	05000 317	1 OF 07

TITLE (4)
Excessive Corrosion of Incore Instrumentation Flange Components

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBERS(S)
										05000
02	21	94	94	-- 004 --	01	07	29	94		05000

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

OPERATING MODE (9)	5		20.402(b)		20.405(c)	50.73(a)(2)(iv)		73.71(b)
			20.405(a)(1)(i)		50.36(c)(1)	50.73(a)(2)(v)		73.71(c)
POWER LEVEL (10)	0		20.405(a)(1)(ii)		50.36(c)(2)	50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iii)		50.73(a)(2)(i)	50.73(a)(2)(viii)(A)		
			20.405(a)(1)(iv)	X	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)	50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Mike D. Milbradt, Compliance Engineer	410-260-4352

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

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	AB	PSF		Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

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

On February 21, 1994, we discovered higher than anticipated corrosion of three nuts on one of the Incore Instrumentation flanges on the Unit 1 reactor vessel head. A subsequent inspection discovered an additional flange with similar degradation. The flanges were known to be leaking slightly since 1993, but we deferred repairs until 1994 because the expected corrosion rate was very low. The excessive corrosion rate was apparently due to the presence of wet boric acid on some of the flange components where we expected only dry boric acid. There were no actual safety consequences, although the potential existed for a significant leak. Corrective action to repair the leaking flanges is complete.

I. DESCRIPTION OF EVENT

On February 21, 1994 during a refueling shutdown inspection of the Unit 1 reactor vessel head area, we discovered higher than anticipated corrosion of three nuts on one of the Incore Instrumentation (ICI) flanges. A subsequent inspection discovered an additional flange with similar degradation. One nut had failed and two adjacent nuts were significantly corroded but had not failed. The corrosion was caused by reactor coolant leakage past the ICI detector joint assembly, leading to a build-up of concentrated boric acid on flange components.

In March, 1993, evidence of boric acid leakage was discovered on the Unit 2 ICI flanges during refueling shutdown inspections. The issue was documented via our Issue Report system and a Root Cause Analysis initiated to determine appropriate corrective actions. An operability evaluation was performed justifying the continued operation of Unit 1. The evaluation included calculation of the potential amount of wastage of the carbon steel nuts and studs, which was deemed to present an acceptably low risk. In April, 1993, the root cause analysis on the Unit 2 leakage was completed and a modification to the Unit 2 ICI flanges initiated. The cause of the leakage was determined to be a change of gasket material in 1988 which changed the crush characteristics of the gasket and allowed leakage past the ICI detector joint. The corrective modification included replacement of the gaskets with a thicker version and the placement of Belleville washers underneath the castle nut assembly to help maintain tension on the joint. This modification was implemented on Unit 2 prior to restart following the 1993 refueling outage. Thus far, we have observed no evidence of any additional leakage from the Unit 2 ICI flanges.

In June, 1993, following indications of increasing Unit 1 containment ambient temperature and decreasing performance of the containment air coolers (CACs), the Plant General Manager directed shutdown of the unit to investigate. During this shutdown, the ICI flanges were examined and seven of eight were found to be leaking. The amount of boric acid leakage was similar to the previously identified leakage on Unit 2. A team was assembled to address the issues and form recovery plans. On June 25, 1993, the Plant Operations and Safety Review Committee reviewed the operability determination for Unit 1. This evaluation had incorporated the following considerations: Technical Specification and Updated Final Safety Analysis Report reviews, evaluation of the effects of boric acid on other containment components and structures, effects of boric acid on the reactor vessel head, effects and extent of boric acid corrosion of the ICI flange studs and nuts, and the potential effects of a fuel failure in conjunction with the leaks. The evaluation concluded that continued operation of the unit was justified. The Plant Operations and Safety Review Committee recommended resumption of full power operation with repair to the flanges deferred until the scheduled 1994 outage. On June 28, 1993, following completion of maintenance activities and a careful review of the issue, the Plant General Manager directed that Unit 1 be returned to service.

II. CAUSE OF EVENT

The apparent cause of the event was an increased presence of wet boric acid rather than dry boric acid on a few of the affected ICI flange stud nuts.

A Root Cause Analysis (RCA) was performed to determine why the corrosion rate was greater than expected. The results of the RCA indicate there were four main factors that contributed to the cause of the event: (1) Use of carbon steel studs/nuts; (2) Incorrect assumption of the ICI flange temperature; (3) Boric Acid Inspection Program weakness; and (4) Failure to use as-found information in the ICI flange stud stress calculations.

The ICI flange studs/nuts were changed from stainless steel to carbon steel in 1987 after it was determined the stainless steel did not meet the impact toughness requirements of the code. Additionally, corrosion resistant nickel plating was not used because it was considered non-structural and could scratch easily during installation. Thus, carbon studs which were susceptible to degradation once the leak started were used.

After discovering the leaks on Unit 2 in 1993, it was assumed the leaking boric acid was dry and a corresponding corrosion rate of 1.6 mils/month was applied. This assumption was made based on an estimated flange temperature of approximately 500 degree Fahrenheit (F) which would have boiled any moisture off. This estimated temperature was stated in the operability evaluation prepared for Unit 1 in 1993 without an accompanying justification for the stated assumption. Actual temperatures around the flange, as measured during Unit 1's startup this Spring, were between 160 degree F and 295 degree F. Because the actual temperatures were lower than assumed, the studs/nuts were exposed to a combination of wet and dry boric acid resulting in a corrosion rate higher than 1.6 mils/month.

As required by Generic Letter 88-05, a Boric Acid Corrosion Inspection Program existed at the time of the event but only required specific inspections for leaks at the beginning and end of each outage. The program did not address leaks discovered outside of the normal inspections. An evaluation of the studs/nuts actual condition would have revealed a higher than estimated corrosion rate due to wet versus dry boric acid.

The only information considered pertaining to the condition of the studs/nuts, was based on the observations from the mechanics who disassembled Unit 2's flanges. The mechanics stated the studs on Unit 2 were not corroded and the nut corrosion was less than 1/2 inch of total material in the worst case. MTEU concluded the amount of wastage was consistent with a dry boric acid corrosion rate and thus used the rate of 1.6 mils/month found in GL 88-05. The use of 1.6 mils/month yielded a calculated stud stress increase of 8.2% The 1/2 inch loss should have been considered taken on a diameter over a 20 month operating cycle which would have equated to a corrosion rate of 12.5 mils/month. This corrosion rate corresponds to a mixture of wet and dry

boric acid corrosion rates. This yields a calculated stud stress increase of 108%. 108% would not have been considered acceptable.

III. ANALYSIS OF EVENT

Each of the eight ICI flange assemblies consists of an upper and lower half secured together with eight 13 inch long 1-3/4 inch diameter studs secured by a 1-3/4 inch heavy hex nut (see Figures). The nuts are made of ASTM A 194 Grade 2H carbon steel and the studs are made of ASTM A 193 Grade B7 carbon steel. In this specific case, three adjacent studs were degraded: one was failed, and two were significantly corroded. An engineering analysis was performed by an engineering consulting firm to investigate the failure sequence of the ICI assembly, and to consider the possibility of an undetected progression of stud corrosion until catastrophic failure. The analysis concluded the ICI flange would first start to open when approximately two adjacent studs fail. Leakage would start to occur when half of a third stud is degraded and would be approximately 16 gpm. With three failed studs, the maximum stress on the remaining studs is below yield stress. Thus, with a leak of 16 gpm, there is no catastrophic failure of the ICI assembly. This level of leakage is well within the capacity of the charging pumps and would be apparent to operators, allowing them to take mitigating actions and safely shutdown the reactor. In the unlikely event of catastrophic joint failure, the effects would be bounded by the Calvert Cliffs loss-of-coolant accident analyses.

Actual leakage in this case was within Technical Specification limits. We experienced no loss of pressure boundary integrity. However, the initial analysis combined with our preliminary assessment of the amount of corrosion we experienced led us to conclude on March 25, 1994 that the potential had existed for a significant leak. We are therefore reporting this event under 10 CFR 50.73(a)(2)(ii).

IV. CORRECTIVE ACTIONS

Corrective action for the ICI flange leakage problem was to implement the modification discussed above. The modification was implemented on Unit 2 in 1993 and thus far we have observed no evidence of further leakage on that unit. In December, 1993, we performed a mock-up test that validated the new design. The modification was implemented on Unit 1 during the 1994 Spring Refueling Outage. Based on the results of the RCA, the following corrective actions are being taken:

A. Operability evaluations will require documented justifications for assumptions to be stated to ensure they are reasonable;

B. The Boric Acid Corrosion Inspection Program will be revised to ensure all boric acid leaks are evaluated;

C. Corrosion resistant studs and nuts will be installed in the ICI flanges during the next refueling outages for both units. The use of corrosion resistant bolting as suggested in GL 88-05 will also be evaluated for future modifications to valve/flange joints on systems with borated water; and,

D. The results of the RCA, purpose of the Boric Acid Corrosion Inspection Program, and importance of using conservative and as-found data when preparing engineering evaluations will be incorporated into staff training.

V. ADDITIONAL INFORMATION

A. Identification of components referred to in this LER:

Component	IEEE 803 EIIS Funct	IEEE 805 System ID
ICI Flange	PSF	AB
ICI Flange Stud	PSF	AB
ICI Flange Hex Nut	PSF	AB
ICI Gasket	PSF	AB

B. There have been no previous similar events.

LICENSEE EVENT REPORT (LER)

TEXT CONTINUATION

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

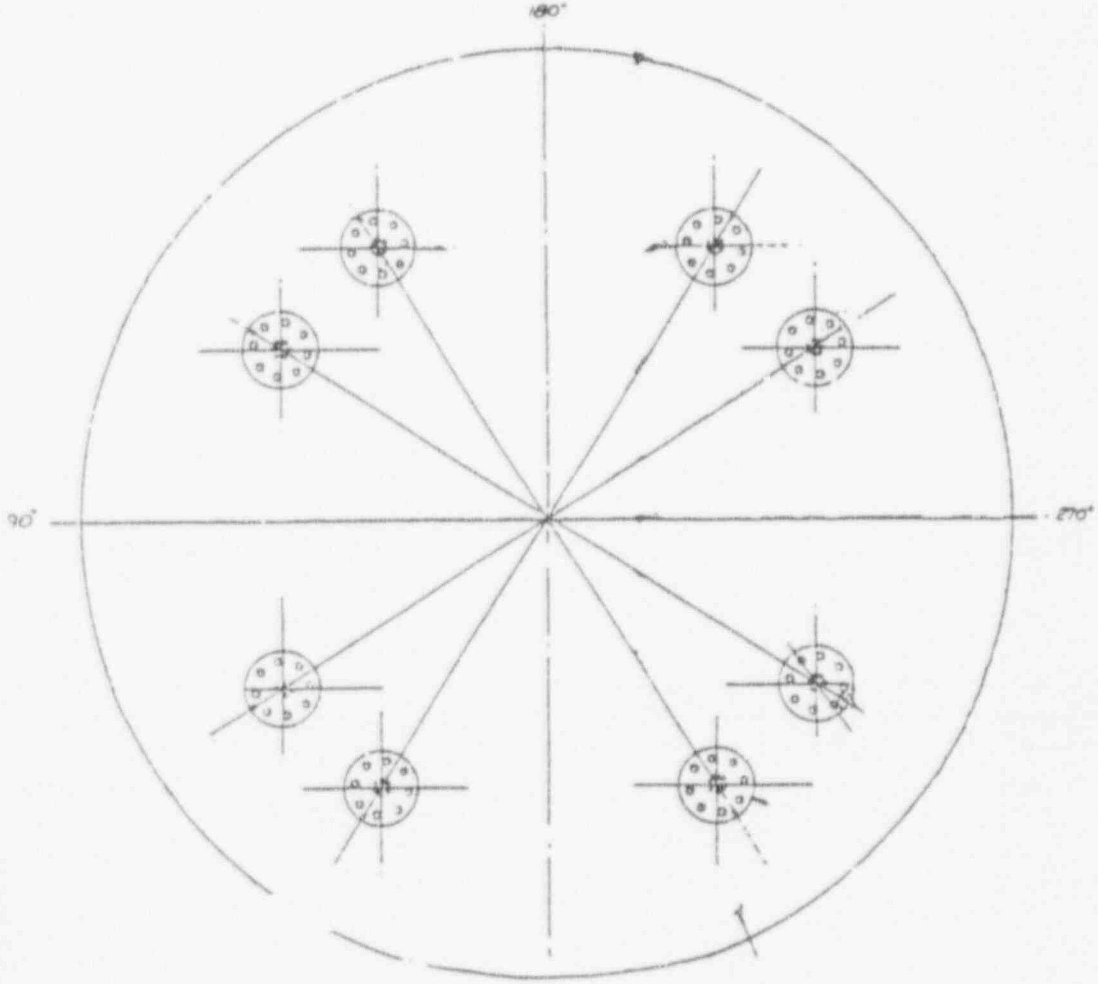

FIGURE 1

PLAN VIEW INDICATING LOCATION AND ORIENTATION
OF ICI FLANGES ON PRESSURE VESSEL CLOSURE HEAD

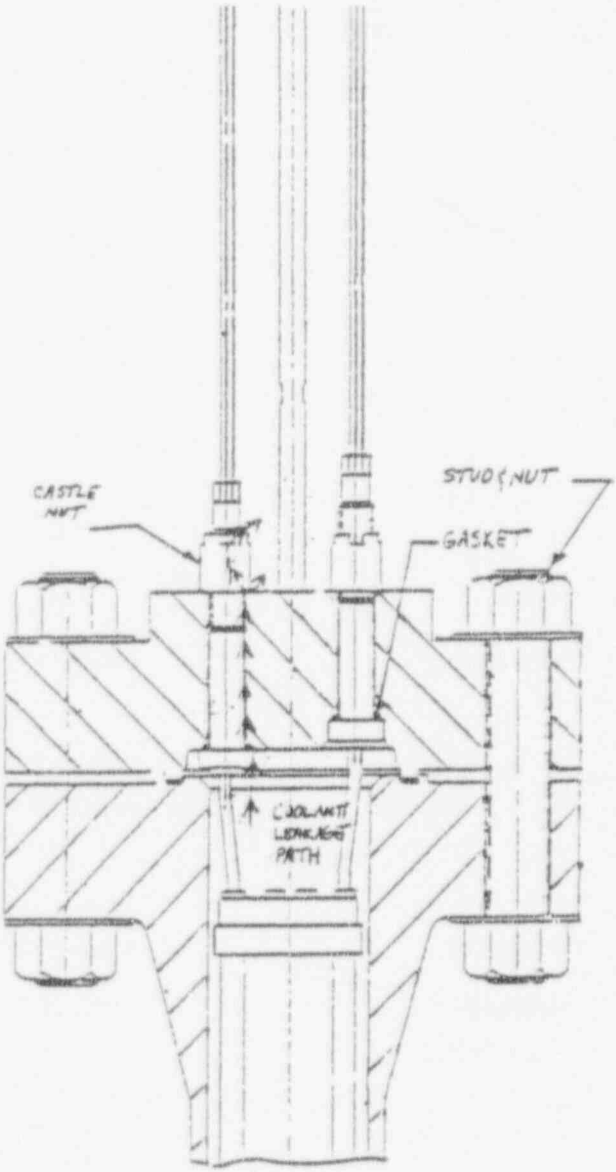

FIGURE 2

ICI FLANGE ASSEMBLY