NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Calvert Cliffs, Unit 1	0 5 0 0 0 3 1 7	1 OF 0 4

TITLE (4)
Main Steam Piping Flaw

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 NUMBER(S)
1 2	0 3	8 6	8 7	— 0 0 2 —	0 0	0 2	0 5	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 0

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

	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)
	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	
	AREA CODE	
Michael J. Gahan, III, Senior Engineer	3 0 1	2 6 0 - 4 4 1 6

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

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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X YES (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	3 0	8 8

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

During refueling MODE 6, while conducting a non-mandatory inspection, a section of thin wall was found on #12 Main Steam Line, adjacent to the second elbow, downstream of the steam generator, inside Containment. Readings as low as 0.86" were found. ANSI B31.1 allowable minimum wall is 0.95". Readings below this were found approximately 22% around the circumference in a band approximately 0.5" wide adjacent the elbow to the pipe weld on the pipe side.

The probable cause was grinding on the edge of the pipe to achieve proper fit-up for welding during initial construction. There is no evidence of erosion, corrosion, nor other active mechanism of degradation. There is no significant safety hazard.

Several analyses (primary stress, fracture mechanics, and finite element) were performed to ensure the effected piping will meet all applicable requirements of ASME Code, Section XI.

Interim approval was received from the NRC on December 19, 1986. A final submittal will be made to the NRC ensuring conformance with Code allowable primary stress levels by the end of the next Unit 1 Refueling Outage (second quarter 1988.)

This LER is being submitted more than thirty days after the event because BG&E had assumed all reporting requirements were met with previous phone conversations, meetings, and submittals that are documented later in the LER. In a meeting on January 9, 1987, we were informed by our Resident Inspector that an LER is required under 10 CFR 50.73(a)(2)(ii)(B). This LER was initiated following that meeting.

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As part of an extensive non-mandatory program to investigate piping for erosion/corrosion effects, ultrasonic thickness measurements were taken on #12 Steam Generator Main Steam Line (EB-01-1005) at the second elbow, downstream from the flow restrictor. The Plant was in Refueling MODE 6 at the time. Initial readings were taken on a grid spacing of 3" by 3". This elbow is inside the Containment at elevation 61'-0". The pipe diameter is 34", the material is ASTM A-155, Grade KC70, Class I, and the design minimum wall thickness is 0.95".

Downstream of the field girth butt weld joining the elbow to the horizontal pipe, reduced wall thickness readings were recorded on the bottom side of the pipe. Readings below 0.95", to as low as 0.86" (90.5% of min. wall), were found immediately adjacent to the weld in a 1/2" band 24" long. This is 22.5% of the circumference. The 1/2" band was defined and confirmed during a second ultrasonic investigation. The wall thickness for the rest of the pipe examined ranged from 1.00" to 1.12".

The probable cause was grinding of the edge of the pipe to achieve proper fit-up for welding during initial construction. Seamed piping normally is difficult to fit-up and some grinding may be required to provide a smooth continuous surface between pipe sections. There is no evidence of erosion, corrosion, nor other active mechanism of degradation. There is no significant safety hazard.

We reviewed the radiograph produced during plant construction. The film showed a darker band in the same area as indicated by the ultrasonic testing. Radiography was repeated on December 3, and the films showed no significant differences. We believe this condition has existed since construction and is essentially in its same shape and size after more than eleven years of operation.

Five other locations on the main steam line were ultrasonically tested to determine if similar conditions existed. Pipe wall thicknesses were all found to be satisfactory.

Nuclear Engineering Services Department (NESD) was notified of the finding on November 26, 1986. A presentation was made to the Plant Operating Safety Review Committee (POSRC) to report the finding and to describe the engineering evaluations that would be performed.

The NRC Resident Inspector was notified by telephone of the condition at 0930 on December 1, 1986. Later that day, M. McBrearty from Region 1 spoke with our representative from Design Engineering. On December 4, 1986, a conference call was held among NRC Region 1 members, NRR, the Resident Inspector, and various members of NESD.

On December 3, 1986, the POSRC was given a full description of the evaluations performed and the recommendation to accept as is. POSRC accepted the recommendation.

On December 10, 1986, a presentation was made to NRR and on December 11, a teleconference was held with NRR to discuss the situation, subsequent actions, and reporting requirements. NESD pursued the provisions for an alternate analysis, as allowed by ASME Code, Section XI, 1974 Edition with Addenda through Summer 1975, IWB-3600. A linear elastic fracture mechanics analysis based on Appendix A and an elastic-plastic/plastic fracture mechanics analysis similar to Article IWB 3640 (1983 through Winter 1985 Addenda) were performed. The critical flaw size was determined for the maximum load condition factoring in design bases transients. Flaw propagation was determined for both a 24 month period and end of life. This analysis determined that there is an adequate margin of safety consistent with ASME Section XI criteria.

A second analysis was performed to determine if the primary stress levels as required by ASME Code Section XI, 1983 Edition with Addenda through Summer 1983, IWB 3610(b) are satisfied (*). The results were evaluated in light of the requirements in the original construction code, ANSI B31.1-1967.

All primary longitudinal stress levels are met for each load combination. The primary hoop stress levels do not meet B31.1, Paragraph 104.1. The analyses as documented demonstrates the following:

- o The as found wall thickness of the the piping can withstand up to hot standby conditions.
- o Short term transient capabilities are within the worst case pressure load for all Chapter 14 analyses.
- o There is a 21% excess reinforcement available in the surrounding pipe wall to ensure structural integrity of the piping system.

In addition, the maximum dynamic hoop stress based on the pressure surge created by closure of the main turbine stop valves combined with design pressure is within the allowable yield stress.

* Our Code of record is the 1974 Edition with Addenda through Summer 1975. The 1983 Edition with Addenda through Summer 1983 will be effective April 1, 1987.

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

On December 16, 1986, a letter from our Mr. A. R. Thornton, General-Supervisor Plants and Project Engineering, to Mr. Ashok C. Thadani, Director, Division of PWR Licensing, NRC, was written to request approval of Main Steam Piping Evaluation per ASME Section XI, IWB-3600 and relief from IWB-3610 (b).

On December 19, 1986, a second letter was written, at the request of the NRC, to request approval to update to ASME Code, Section XI, 1983 Edition through Summer 1983 for #12 Steam Generator Main Steam Line (EB-1-1005) at the second elbow downstream from the flow restrictor adjacent to and downstream of ISI weld number 34-MS-1205-8.

On December 19, 1986, a letter from Mr. Frank J. Miraglia, Director, Division of PWR Licensing-B to Mr. J. A.Tiernan, Vice-President, Nuclear Energy, was written to grant interim approval of ASME Code update and relief for main steam piping flaw at Calvert Cliffs Unit 1. The Commission understands that BG&E has committed to ensuring that the effected piping will meet all applicable requirements of ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda prior to entering MODE 2 following the next Unit 1 Refueling Outage (second quarter 1983).

On December 23, 1986, a draft report "Stress Analysis of This Pipe Region in #12 Steam Generator Main Steam Line (EB-01-1005-05)" prepared by Southwest Research, Inc. discussed the results of finite element analysis which predicted a hoop stress level of 15,874 psi. This is below the code allowable of 17,500 psi.

The Southwest Research, Inc. report will be finalized and submitted to the NRC in a letter demonstrating that the pipe meets all applicable requirements of ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda.

Throughout the process, the consensus was that the existing condition poses no significant safety hazard.

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BALTIMORE GAS AND ELECTRIC

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NUCLEAR OPERATIONS DEPARTMENT
CALVERT CLIFFS NUCLEAR POWER PLANT
LUSBY, MARYLAND 20657

February 5, 1987

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

Docket No. 50-317
License No. DPR 53

Dear Sirs:

The attached LER 87-02 is being sent to you as required by 10 CFR 50.73.

Should you have any questions regarding this report, we would be pleased to discuss them with you.

Very truly yours,

J R Lemons

J. R. Lemons
Manager - Nuclear Operations Department

MJG
JRL:MJG:pah

cc: Dr. Thomas E. Murley
Director, Office of Management Information
and Program Control
Messrs: J. A. Tiernan
W. J. Lippold

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