NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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TITLE (4)
Reactor Recirculation B Loop, High Pressure Flow Element Venturi Instrument Line Steam Leakage Results in Unit 3 Shutdown Due to Fatigue Failure of Socket Welded Pipe Joint

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 NAME	DOCKET NUMBER
									N/A	N/A
03	21	99	99	003	00	04	20	99	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	x	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	038		20.2203(a)(2)(i)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.405(a)(1)(ii)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Safwat Gibraiel, Design Engineer	(815) 942-2920 ext 3121

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		06	15	99

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

On March 21, 1999, at approximately 0900 hours, during an inspection of the drywell for the source of a previously detected increase in unidentified leakage, a steam leak was discovered on a socket weld at a 1 inch tee fitting on the high pressure instrument line for the Reactor Recirculation (recirc.) pump Loop B venturi flow element. A Unit 3 shutdown was performed in accordance with the pre-approved inspection plan, as required by Technical Specification 3.6.H for primary pressure boundary leakage.

Preliminary analysis of the failed tee weld indicates the failure can be attributed to fatigue cracking (most probably from vibration), initiating from defects due to lack of weld fusion in the root of the socket weld. The failed tee was removed and will have a detailed material analysis performed to determine the mode of failure. This same tee weld had previously failed in November 1997 (ref. LER 97-012-00). No detailed material analysis was performed after that failure. A root cause investigation will be performed to determine the reasons the corrective actions implemented after the November 1997 weld failure were deficient.

The overall safety significance of the event was minimal.

This report is submitted pursuant to 10 CFR 50.73(a)(2)(i)(A).

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2527 MWt rated core thermal power

Energy Industry Identification System (EIIS) Codes are identified in the text as [XX] and are obtained from IEEE Standard 805-1984, IEEE Recommended Practice for System Identification in Nuclear Power Plants and Related Facilities.

EVENT IDENTIFICATION:

Reactor Recirculation B Loop, High Pressure Flow Element Venturi Instrument Line Steam Leakage Results in Unit 3 Shutdown Due to Fatigue Failure of Socket Welded Pipe Joint

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 3 | Event Date: 3/21/99 | Event Time: 1138 CST

Reactor Mode: 1 | Mode Name: Run | Power Level: 38 percent

Reactor Coolant System Pressure: 900 psig

B. DESCRIPTION OF EVENT:

This report is being submitted in accordance with 10CFR50.73(a)(2)(i)(A), which requires reporting the completion of any nuclear plant shutdown required by the plant's Technical Specifications (TS).

On March 13, 1999 a drywell Continuous Air Monitoring (CAM)[IK] system activity alarm was received in the Control Room, indicating an increase in unidentified leakage into the drywell [BD]. The activity level subsequently stabilized. On March 20, 1999 the drywell activity level again began increasing. Management conservatively decided to make a planned entry into the drywell at power to locate and repair the source of the unidentified leakage.

On March 21, 1999, at approximately 0900 hours (during an inspection of the drywell for the source of the increase in unidentified leakage), a steam leak was discovered on a socket weld at a 1 inch tee fitting on the high pressure instrument line for the Loop B recirc. pump venturi flow element. The reactor was operating at approximately 39 percent power at the time of discovery.

Following prompt assessment of the steam leak, a shutdown of Unit 3 was performed. This was in accordance with T/S 3.6.H because the leakage was from the primary pressure boundary. Unit shutdown commenced on March 21, 1999 at approximately 1115 CST. ENS notification was made at 1143 CST and cold shutdown was achieved on March 21, 1999 at 1920 CST.

The tee and nearby piping were replaced in 1986 as a result of the Reactor Recirc. piping replacement modification, M12-3-85-16. The weld on the same tee was repaired on November 5, 1997 as a result of a leak similar to the most recent leak.

The instrument piping is stainless steel and originates at two locations on the B recirc. loop venturi, high-pressure side. Two branch pipes route around the recirc. piping and join together at the subject tee fitting. The piping is then routed through drywell penetration X-131C to flow transmitters 3-261-6C & 6D (see sketch on page 6).

There are two guide-type supports between the tee and the drywell penetration. The supports are closer to the drywell penetration than to the recirc. loop piping. Those two supports are U-bolts, with gaps, intended to act as two-way restraints.

The steam leak was located in the top portion of the pipe-to-tee socket weld connection. A Management decision was made to replace the piping, two elbows, and the tee with new material. Stronger welds

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replaced the socket welds in this portion of the line. The flawed piece will have a detailed material analysis performed to determine the failure mode, which will help in determining the root cause of this failure. This information will be made available upon completion of the root cause investigation and will be documented in a supplement to this LER.

The piping replacement was successfully completed and tested, with final visual inspection on March 23, 1999.

No other system or component being out of service contributed to the event.

C. CAUSE OF EVENT:

The cause of this event cannot be definitively determined until the final material analysis results have been reviewed. However, preliminary analysis results for the failed tee weld indicate the failure can be attributed to fatigue, initiating from a defect due to lack of weld fusion in the root of the socket weld. This preliminary result is supported by industry research. EPRI report TR-107455 (Vibration Fatigue of Small Bore Socket-Welded Pipe Joint) and O&MR 01-98 (Small Bore Piping Connection Failures – Operation and Maintenance Reminder 424) cite examples where this type of failure can be attributed to fatigue due to vibration in conjunction with a pre-existing flaw (with the potential of poor weld and/or poor socket-pipe fit-up). The vibration in the line works on a flaw in the root weld. When sufficient cycles are reached, the weld fails with crack initiation on the inside of the weld.

The long term corrective action from the 1997 weld failure recommended a PT examination. This type of examination will indicate presence of flaw at the outside of the weld but will not give any indication of a flaw on the inside of the weld (which is the most common cause of socket weld failure). Therefore, a PT examination is not adequate to detect this type of weld failure.

Also, the long-term corrective action from the previous failure did not require supporting the pipe to reduce vibration due to the potential of thermally over-stressing the piping. However, the instrument line could be supported from the large recirculation pipe, which would eliminate the thermal over-stress concern since both pipes move together (tieback support).

Another recommendation by the EPRI study is to improve the fatigue strength of the weld by increasing the weld leg length along the pipe (to twice the socket side of the weld). The corrective actions taken after the November 1997 failure did not include changing the size of the weld.

D. SAFETY ANALYSIS

The consequences of this incident had minimal impact on reactor safety. The actual leak was detected by the drywell leak monitoring systems well in advance of any serious degradation to the primary boundary. The leak monitoring system was able to identify the leak at a level of 0.25 gpm increase to the floor drain sump collection. Necessary inspections were scheduled which identified the specific leaking weld and a controlled shutdown was initiated to affect repairs.

Further evidence that the Reactor Recirculation instrument line pipe leak had minimal safety significance is the fact that had the instrument line failed completely, a reactor scram would have occurred. This is because the flow element signal associated with the sensing line inputs the flow bias scram setting for the Average Power Range Monitors (APRM)[IG]. Since the leak was found on the high side of the flow element sensing line, the scram value for the flow biased setting would have been reduced significantly below 100 percent power, thus resulting in a reactor scram.

Finally, a catastrophic failure of the instrument sensing line is bounded by the analyzed condition of a small break LOCA. This postulated failure would cause a drywell high-pressure signal to initiate the Emergency

Core Cooling Systems (ECCS) and a reactor scram signal. The consequence of this accident would be mitigated by the HPCI System or the Automatic Depressurization System (ADS) [SB] in conjunction with the Low Pressure Coolant Injection System (LPCI) [BO] and Core Spray [BM] system.

Therefore, the safety significance of the event is considered to be minimal.

E. CORRECTIVE ACTIONS:

The tee with the failed weld, the two adjacent elbows (to the left and right sides of the run element of the tee), the pipe between the elbows, and approximately 5 inches of the pipe welded to the branch side of the tee were replaced with type 304 stainless steel. The original material was type 316L stainless steel, which has a lower stress allowable. (complete)

EPRI report TR-107455 states, "Most significantly, increasing the axial weld leg dimension with respect to ASME Code minimums can substantially increase the fatigue strength of the joint and can even counteract the potentially damage effect of weld root defects." In keeping with this recommendation, the old welds were replaced with a stronger weld (the weld leg length along the pipe is twice the socket height). The leg length of the old weld was equal to the socket height. The existing pipes that are welded to the new elbows were polished and non-destructive surface examined by penentrant testing (PT) before welding. (complete)

The above corrective actions result in a much stronger configuration than the originally installed design (which lasted for 11 years before the first failure in 1997). Also, the new weld was made in the shop where the conditions are much better than in the field (the failed weld is very close to the recirc. pipe header, which may have made it difficult to perform the field weld in 1997). For this reason, the station has a high degree of confidence that the piping will have an installed life at least comparable to the originally installed design.

Prior to the fitting replacement, the high and low pressure flow element sensing lines on both Unit 3 recirc. loops were visually inspected and no noticeable damage or deformation in the pipe or supports were observed. (complete)

The removed tee and elbow fittings (including approximately one inch of piping welded to all sides of the tee and one side of each elbow) were sent to Argonne National Laboratory for further analysis to determine the root cause of the failure. The analysis results are expected in May 1999. The corrective actions taken after the 1997 weld failure have proven to be inadequate or inappropriate. An investigation will be performed to determine the root cause of that deficiency. The results of this investigation and the results of the analysis being performed by Argonne National Laboratory will be submitted as a supplement to this LER. (NTS#: 2491809900301)

The Unit 3 Loop B venturi flow element high-pressure instrument piping analysis will be evaluated. Depending on the results of the root cause analysis of the failed weld and adjacent, the piping systems (for all units and loops) may be modified to reduce vibration in the instrument line. (NTS#: 2491809900302)

F. PREVIOUS OCCURRENCES:

On November 1, 1997, at approximately 1130, during an inspection of the Unit 3 drywell for the source of a previously detected increase in unidentified leakage, a steam leak was discovered on a socket weld at a one inch tee fitting in the high pressure instrument line for the Reactor Recirculation (Recirc.) pump Loop B venturi flow element. The reactor was operating at approximately 39 percent power at the time of discovery. A Unit 3 shutdown was promptly performed in accordance with the pre-approved inspection plan, as required by Technical Specification 3.6.H for primary pressure boundary leakage.

The cause of the event was attributed to fatigue failure of the subject socket weld. It was postulated that a flaw was present in the socket root weld, which led to a premature failure of the weld in the presence of vibration.

Corrective actions included: the failed weld was replaced; adjacent welds were non-destructive surface examined by penentrant testing (no indications found); and, the instrument line piping analysis was to be reviewed to determine if improvements should be made to reduce vibration in the line. The overall safety significance of the event was minimal.

After review of the instrument line piping analysis, it was determined that the most prudent action to address this issue is to regularly PT the affected piping as part of the refuel outage activities. The welds at the tee and elbows for loops A and B were PT examined during the D3R15 outage and no indications were found. These corrective actions were ineffective to prevent recurrence.

G. COMPONENT FAILURE DATA:

None.

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BILL OF MATERIALS					
ITEM	QUANTITY	LINE-SIZE	SCHEDULE	MATERIAL	DESCRIPTION
1	1	1" NPS	80S	SA312 TP316L	S.S. PIPE, 4'-0" LONG
2	1	1" NPS	--	SA403 or SA182 WP316L	3000# S.W. 1"X1" TEE
3	1	1" NPS	80S	SA312 TP316L	S.S. PIPE, 6'-8" LONG
4	1	1" NPS	80S	SA312 TP316L	S.S. PIPE, 7'-6" LONG
5	1	1" NPS	--	SA403 or SA182 WP316L	3000# S.W. 90° ELBOW
6	1	1" NPS	80S	SA312 TP316L	S.S. PIPE, 5'-8" LONG
7	1	1" NPS	--	SA403 or SA182 WP316L	3000# S.W. COUPLING
8	4	1" NPS	--	SA403 or SA182 WP316L	3000# S.W. 90° ELBOW (SEE NOTE 7)