NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 06/30/2001

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TITLE (4)
Supplement to 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	1999	1999	003	01	08	30	1999	FACILITY NAME N/A	DOCKET NUMBER N/A

OPERATING MODE (9)	1
POWER LEVEL (10)	38

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)
	20.2203(a)(i)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i))		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)
Frank P. Polak, Rapid Response Team Design Engineer	(815) 942-2920 ext 2831

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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-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 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.

A detailed analysis of the failed tee weld indicated the failure was attributed to vibration induced fatigue cracking accelerated by the presence of a weld defect due to lack of fusion in the root of the socket weld. 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 has been performed to determine the reasons the corrective actions implemented after the November 1997 weld failure did not prevent recurrence.

The overall safety significance of the event was minimal.

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

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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:

Supplement to 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: 03-21-99 Event Time: 1138
Reactor Mode: 1 Mode Name: Run Power Level: 038
Reactor Coolant System Pressure: 0900 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, an increase in the drywell Continuous Air Monitoring (CAM)[IK] system activity indicated a potential for 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 recirculation pump venturi flow element. The reactor was operating at approximately 38 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 Recirculation 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 recirculation loop venturi, high-pressure side. Two branch pipes route around the recirculation 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 recirculation 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. The new socket welds

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were installed per the recommendations of ERPI TR-107455, Vibration Fatigue of Small Bore Socket-Welded Pipe Joints. The socket weld piping axial leg length with respect to the fitting shoulder height was installed with a 2 to 1 leg length ratio respectively. Adding a last pass on the pipe side of the weld is referred to as 'last pass improvement'. The last pass improvement welding technique, in comparison to standard ASME Code minimum socket welds, enhances the joints resistance to high cycle fatigue. The piping replacement was successfully completed and tested, with final visual inspection on March 23, 1999.

No other systems, components or structures were identified which contributed to the isolation event.

C. CAUSE OF EVENT:

A detailed analysis for the failed tee weld indicated the root cause of the weld failure was attributed to fatigue crack propagation resulting from low amplitude, high cycle vibration which was accelerated by the presence of a lack of fusion root weld defect. This conclusion is supported by industry research. EPRI Technical Report (TR) 107455 (Vibration Fatigue of Small Bore Socket-Welded Pipe Joint) and INPO Operations and Maintenance Reminder (O&MR) 01-98 (Small Bore Piping Connection Failures – Operation and Maintenance Reminder 424) cite examples where this type of failure is attributed to fatigue due to vibration in conjunction with a pre-existing flaw. 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.

A Root Cause Report (RCR) was performed to determine why the corrective actions implemented due to the November 1997 event were ineffective in preventing recurrence. The root cause of the ineffective corrective actions is human performance. The following deficiencies in human performance were noted:
(NRC Cause Code A)

1. Enforcement of Engineering Department Standards by the Lead Design Engineer was less than adequate. This resulted in the failure:

   a) of engineering personnel to perform industry research on vibration fatigue of small bore socket-welded pipe joints;

   b) to require a review of the corrective actions to prevent recurrence (NTS# 2491809701201) by site or corporate welding Subject Matter Experts (SMEs);

   c) to challenge engineering personnel to aggressively identify the root cause and implement effective corrective actions to prevent recurrence.

2. Implementation of Engineering Department Standards by the Design Engineering corrective action owner were less than adequate. This resulted in the failure of the corrective actions owner:

   a) to read LER 97-012 which included the summary of EPRI Report TR-107455, Vibration Fatigue of Small Bore Socket-Welded Pipe Joints;

   b) to elevate the development of the corrective actions beyond a long term, low priority administrative task.

Contributing Causes include:

1) A perception by upper station management the November 1997 failure was an isolated occurrence. This incorrect conclusion resulted in the decision to perform a repair verses a replacement and subsequent detailed analysis of the affected material.

2) The author of LER 3-97-012, after identification of EPRI Report TR-107455, Vibration Fatigue of Small Bore Socket-Welded Pipe Joints, failed to require a documented engineering review of the report per NTS.

3) Engineering knowledge based deficiency on how socket welds tend to fail due to high-cycle/low amplitude vibration.

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 complete the 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 corrective actions to improve the fatigue life of the socket welded fitting include:

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)

Prior to the fitting replacement, the high and low pressure flow element sensing lines on both Unit 3 recirculation loops were visually inspected and no noticeable damage or deformation in the pipe or supports were observed. (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 weld more resistant to a vibration induced fatigue failure (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 penetrant testing (PT) before welding. (Complete)

The above corrective actions resulted in a configuration which is less susceptible to vibration induced fatigue 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, which reduces the

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probability of a socket weld root defect. For these reasons, the station has a high degree of confidence that the piping will have an improved resistance to vibration induced fatigue in comparison to the originally installed design.

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. The results of the analysis are included in this LER Supplement. The corrective actions taken after the 1997 weld failure have proven to be inadequate or inappropriate. The results of the Root Cause Report are included in this LER Supplement.

The corrective action to prevent recurrence include:

1)) Counsel the appropriate engineering personnel on the implementation of Engineering Department Standards, including the use of Subject Matter Experts and implementation of industry experience. (Complete).

2) Train engineering personnel on the performance of electronic industry data searches. Training completed during Engineering Personnel Support Training (ESPT). (Complete).

3) Review this event with engineering personnel and reinforce Engineering Department Standards. Training completed during ESPT. (Complete).

4) Design Engineering shall issue a Design Change Package (DCP) to install pipe supports on the high and low pressure sensing lines for the Unit 2(3) Reactor Recirculation A (B) Loops. (AR# 7354-07).

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 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 penetrant 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	--	[illegible]	3000# S.W. 1"X1" TEE
3	1	1" NPS	80S	SA312 TP316L	S.S. PIPE [illegible] LONG
4	1	1" NPS	80S	SA312 TP316L	S.S. PIPE 7'-8" LONG
5	1	1" NPS	--	[illegible]	3000# S.W. 90° ELBOW
6	1	1" NPS	80S	SA312 TP316L	S.S. PIPE [illegible] LONG
7	1	1" NPS	--	[illegible]	3000# S.W. COUPLING
8	4	1" NPS	--	[illegible]	3000# S.W. 90° ELBOW (SEE NOTE 7)