NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), 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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Dresden Nuclear Power Station, Unit 3	05000249	1 OF 4

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 None	DOCKET NUMBER
11	01	97	97	012	00	11	26	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	039

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

	20.2201(b)		20.2203(a)(3)(i)		50.73(a)(2)(iii)		73.71(b)
	20.2203(a)(1)		20.2203(a)(3)(ii)		50.73(a)(2)(iv)		73.71(c)
	20.2203(a)(2)(i)		20.2203(a)(4)		50.73(a)(2)(v)		OTHER
	20.2203(a)(2)(ii)		50.36(c)(1)		50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)
	20.2203(a)(2)(iii)		50.36(c)(2)		50.73(a)(2)(viii)(A)		
	20.2203(a)(2)(iv)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(B)		
	20.2203(a)(2)(v)		50.73(a)(2)(ii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
P. Garrett, Plant Engineering Ext. 2713	(815) 942-2920

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
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On November 1, 1997, at approximately 1130, during 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 in the high pressure instrument line for the Reactor Recirculation (Recirc) pump Loop B Venturi flow element. The reactor was operating at approximately 39% 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 is attributed to fatigue failure of the subject socket weld. It is postulated that a flaw was present in the socket root weld which lead to a premature failure of the weld in the presence of vibration. Corrective actions include: failed weld was replaced, adjacent welds were non-destructive surface examined by penentrant testing with no indications found, and the instrument line piping analysis will be reviewed to determine if improvements should be made to reduce vibration in the line. The overall safety significance of the event was minimal.

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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.

Reactor Recirculation System [AD]

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: 11/01/97 Event Time: 1130

Reactor Mode: 1 Mode Name: Run Power Level: 39%

Reactor Coolant System Pressure: 923 psig

B. DESCRIPTION OF EVENT:

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

On October 17, 1997, 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 increased leakage was confirmed by an increase in Drywell air sample activity. On October 18, 1997, an increase in Drywell floor drain sump flow was identified by Operations personnel. The increase was approximately 0.25 gpm, for a total leakage of 0.83 gpm. Trending of the unidentified leakage continued and contingency plans were developed if leakage increased further. Through Drywell air sampling via the Drywell air monitoring system, the leak was determined to be located in the area of the B Reactor Recirculation (Recirc) pump. On October 29, 1997, with an unidentified leakage increase of approximately 0.45 gpm, Management decided to make an at power entry into the Drywell to locate and repair the source of the unidentified leakage.

On November 1, 1997, at approximately 1130, during 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 in the high pressure instrument line for the Recirc pump Loop B Venturi flow element. The reactor was operating at approximately 39% power at the time of discovery. Following prompt assessment of the steam leak, a Unit 3 shutdown was performed in accordance with the preapproved inspection plan. This was in accordance with TS 3.6.H because the leakage was from the primary pressure boundary.

Unit shutdown commenced on November 1, 1997, at 1331, an ENS notification was performed at 1358, and cold shutdown was achieved on November 2, 1997, at 0328.

The tee and nearby piping were replaced in 1986 as a result of the Reactor Recirc piping replacement modification M12-3-85-16. The subject 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 transmitter 3-261-6C & 6D. There are three guide type supports between the tee and the Drywell penetration. The supports are nearer to the Drywell penetration than to the Recirc loop piping. The first two supports are u-bolts, with gaps, intended to act as two way restraints. It was noted that the piping configuration is relatively flexible near the attachment points to the Recirc loop piping. This is as-designed and allows for the piping to thermally move with the Recirc loop piping. Video camera observations, during the at power Drywell entry, noted that the subject piping vibrated during Recirc loop operation.

The steam leak was located in the top portion of the pipe-to-tee socket weld connection in the center of the weld. A Management decision was made to repair the failed weld only, versus cutting out the entire tee and replacing the tee and all of the welds. This was to minimize the radiation exposure to the workers and the time spent relying on freeze seals to isolate the piping for repairs.

While repairing the weld, it was identified by station maintenance personnel that the as-found pipe socket fit did not have a gap separating the end of the pipe and the seat of the socket. When constructing the socket type joint, it is required to have a minimum 1/16 inch gap between the pipe and the socket seat. The purpose of the gap is to minimize the residual stresses introduced in the weld during the welding process. However, an as-found gap is an indeterminate indication of the actual fit-up gap used at the time of joint construction.

The weld replacement was successfully completed, tested, with final visual inspection by November 5, 1997.

No other system or component inoperabilities have been identified which contributed to the event.

C. CAUSE OF EVENT:

The cause of this event is attributed to fatigue failure of the subject socket weld, NRC Cause Code X. It is 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.

The root weld defect was probably introduced during weld construction. The system vibration then acted on the defect in the root weld, introduced during construction, resulting in premature failure of the joint due to fatigue. Note: these conclusions could not be confirmed through laboratory analysis of the specific failed weld because the failed weld could not be retained due to the repair process utilized.

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 due to the fact that 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% 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 failed weld on the tee socket (loop B high pressure tap line) was replaced. (complete)

The remaining two welds on the subject tee, and two welds on each of the two elbows nearest the subject tee were non-destructive surface examined by penentrant testing (PT). In addition, the three welds on the loop A high pressure tap line tee, and the welds nearer the tee on each of the two nearest elbows was PT. None of the PTs indicated surface flaws in the welds. (complete)

The Unit 3 Loop B Venturi flow element high pressure instrument piping support analysis will be evaluated to determine if improvements should be made to reduce vibration in the instrument line. (2491809701201)

F. PREVIOUS OCCURRENCES:

None.

G. COMPONENT FAILURE DATA:

None.