NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
R.E. Ginna Nuclear Power Plant	05000244	1 OF 7

TITLE (4)

Leak Outside Containment, Due to Weld Defect, Results in Leak Rate Greater Than Program Limit

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
07	23	96	96 --	009 --	02	08	11	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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)(1)		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)
John T. St. Martin - Technical Assistant	(716) 771-3641

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
B	BE	PSF	0000	N					

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 July 23, 1996, at approximately 0857 EDST, it was determined that the leak rate (from a leak on a piping system outside Containment) was greater than the analyzed value of two gallons per hour for a primary coolant source outside Containment.

Immediate corrective action was to conservatively enter Technical Specification Limiting Condition for Operation 3.0.3 and initiate a plant shutdown, and to initiate actions to isolate the leak using a freeze seal. Once the leak was isolated, the plant exited Technical Specification 3.0.3 and the shutdown was stopped. The leaking pipe was cut out and replaced.

The underlying cause of the leak was a weld defect which dates back to original construction in the 1960's.

This event is NUREG-1022 Cause Code (B).

Corrective action to prevent recurrence is outlined in Section V.B. Additional corrective actions are identified in this supplement.

I. PRE-EVENT PLANT CONDITIONS:

On July 22, 1996, the plant was in Mode 1 at approximately 100% steady state reactor power. At approximately 1130 EDST, a Radiation Protection (RP) technician, who was performing a routine weekly survey of the Auxiliary Building, observed water dripping in an area behind the Refueling Water Storage Tank (RWST). The RP technician notified the Control Room operators. Efforts to locate the source of the dripping water and to quantify the leak rate were initiated by Operations and Nuclear Engineering Services (NES). The source of water was determined to be a small leak on a test line for the Containment Spray (CS) system in the Auxiliary Building, which is outside the Containment (CNMT). The leak rate was estimated to be approximately one (1) gallon per hour (GPH).

The Control Room operators did not identify any requirement that would limit plant operations. The Shift Supervisor notified Operations management, maintenance management, and plant and NES staff of the leak. All reached a similar conclusion. The staff worked on developing and implementing an action plan to address the problem. The issues of line integrity and RWST operability were investigated by NES and Laboratory Inspection Services (LIS) personnel. LIS personnel measured the amount of pipe cross-section and weld metal remaining, and NES personnel determined that there was ample metal in the affected area to provide system integrity.

The source of the leak was identified as a small pinhole leak on a pipe-to-90 degree forged socket elbow weld on a two (2) inch nominal pipe size Schedule 10 stainless steel test line for the CS system. The leak was in a part of the CS system that could not be isolated from the RWST by valve manipulations without making both trains of the emergency core cooling system (ECCS) inoperable. The leak was monitored throughout the remainder of the day and night.

An operability assessment had been requested by the Shift Supervisor on July 22, when the existence of the leak had first been identified. On the morning of July 23, 1996, this assessment was reviewed by the Plant Operations Review Committee (PORC). This assessment, which was based on conservative assumptions, concluded that the RWST and ECCS were still operable with the unisolated leak. This conclusion was supported by the fact that there was still ample metal area in the cross-section that could provide integrity of the pipe during a seismic event and that the total combined stress was less than the yield strength of the material.

PORC requested a more accurate estimate of the leak rate. The leak rate was measured and estimated to be 2.6 GPH. This leak rate is in excess of two (2) GPH, which is specified in Surveillance Test Procedure PT-39, "Leakage Evaluation of Primary Coolant Sources Outside Containment", as the program limit for the maximum integrated leak rate outside CNMT from these sources. Since the leak rate exceeded the requirements of procedure PT-39, it was initially judged to be a violation of the "Primary Coolant Sources Outside Containment Program", as specified in the Ginna Station Improved Technical Specifications (ITS) Section 5.5.2. The PORC chairman directed the Shift Supervisor to enter ITS Limiting Condition for Operation (LCO) 3.0.3, based on the conservative requirements of procedure PT-39 and Administrative Procedure A-52.4, "Control of Limiting Conditions for Operating Equipment", even though the plant was in compliance with all ITS LCOs.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	96	-- 009 --	02	3 OF 7

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

II. DESCRIPTION OF EVENT:

A. DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES:

- July 22, 1996, 1130 EDST: A leak is identified on a line outside of Containment.
- July 23, 1996, 0857 EDST: Event Date and Time and Discovery Date and Time.
- July 23, 1996, 0900 EDST: Load reduction is started.
- July 23, 1996, 1440 EDST: Integrated leakage outside Containment is determined to be less than two GPH. Load reduction is stopped.

B. EVENT:

On July 23, 1996, at approximately 0857 EDST, with the plant in Mode 1 at approximately 100% steady state reactor power, the Plant Operations Review Committee (PORC) conservatively directed the Shift Supervisor to enter ITS LCO 3.0.3. The Shift Supervisor directed the Control Room operators to initiate a plant shutdown. At approximately 0900 EDST, the Control Room operators initiated a plant shutdown per Normal Operating Procedure O-2.1, "Normal Shutdown to Hot Shutdown".

The Mechanical Support group initiated actions to provide a freeze seal to isolate the source of leakage from the RWST. A freeze seal was initiated between the leaking socket and the CS pump suction line from the RWST. At approximately 1440 EDST on July 23, 1996, the freeze seal had isolated the leak. ITS LCO 3.0.3 was exited and the load reduction was stopped. The affected pipe and socket welds were cut out and a new prefabricated spoolpiece was installed. The affected weld was retained for failure mode analysis.

A load increase to return the plant to full power was initiated at approximately 1530 EDST on July 23, 1996.

C. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

None

D. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

Due to the location and size of the leak, an operability assessment determined that no emergency safeguards system function was made inoperable. However, the leak was not isolable from the RWST without valve manipulations that would have resulted in the inoperability of the RWST and both trains of the ECCS.

E. METHOD OF DISCOVERY:

The leak was first discovered by an RP technician during a routine weekly survey of the Auxiliary Building. He notified the Control Room operators, and the precise location was confirmed by Operations and Engineering. Confirmation that the leak rate was above 2 GPH occurred after measurements performed at the direction of PORC.

F. OPERATOR ACTION:

When notified by PORC, the Shift Supervisor directed the Control Room operators to enter ITS LCO 3.0.3 and to initiate a plant shutdown. The Control Room operators initiated a plant shutdown per procedure O-2.1.

The Shift Supervisor subsequently notified the NRC per 10 CFR 50.72 (b) (1) (i) (A), non-emergency one hour notification, at approximately 0951 EDST on July 23, 1996. When the leak was isolated, ITS LCO 3.0.3 was exited, and the plant was returned to full power.

G. SAFETY SYSTEM RESPONSES:

None

III. CAUSE OF EVENT:

A. IMMEDIATE CAUSE:

The immediate cause of the condition prohibited by Technical Specifications was the conservative determination by PORC to enter ITS LCO 3.0.3 due to the leak.

B. INTERMEDIATE CAUSE:

The intermediate cause of the leak was a pinhole leak in the weld attaching the pipe to the socket elbow.

C. ROOT CAUSE:

The affected pipe, socket, and fillet weld were analyzed for the failure mode. The pinhole was determined to be an original installation weld defect (dating from the 1960's) that existed over an area of slag inclusion in the weld. Almost all the slag had been leached from the pinhole channel over the past 25 years.

This event is NUREG-1022 Cause Code (B), "Design, Manufacturing, Construction / Installation".

The small pinhole leak does not meet the NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", definition of a "Maintenance Preventable Functional Failure".

IV. ANALYSIS OF EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a) (2) (i) (B), "Any operation or condition prohibited by the plant's Technical Specifications". The integrated leak rate in excess of 2 GPH outside CNMT resulted in conservative entry into ITS LCO 3.0.3. ITS LCO 3.0.3 is only applicable to requirements contained in Chapter 3 of the ITS. There was no license requirement for entry into ITS LCO 3.0.3. Nevertheless, since the plant voluntarily entered ITS LCO 3.0.3, this condition is reportable.

An assessment was performed considering both the safety consequences and implications of this event. The following are the results and conclusions of this assessment:

There were no operational or safety consequences or implications attributed to the leak on the CS test line because:

- Almost all the slag had been leached from the pinhole channel and there was no sign of corrosion or stress (cracking) attack of the stainless steel material adjacent to the pinhole. Since the pinhole did not degrade the cross-sectional area of the forged socket elbow and pipe, the stresses on this line due to the concurrent loads of deadweight, pressure, seismic, and thermal effects were still within the code limit. Seismic integrity was not threatened by the leak nor by the installation of the freeze seal.

- The nature of this leak mechanism would not lead to an increase in hole size above that initially identified.

- The line containing the affected elbow experiences flow from the Refueling Water Storage Tank (RWST) to the Containment Spray (CS) eductor suction valves during periodic surveillance testing and is not subjected to any appreciable dynamic stresses during testing, nor would it be expected to during post-accident operation of the CS or safety injection.

- The offsite exposure due to a large break loss of coolant accident (LOCA) is analyzed in the Ginna Station Updated Final Safety Analysis Report (UFSAR). The subject leak location is exposed to radioactive sump fluid following a small or large break LOCA during recirculation.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	96	-- 009 --	02	6 OF 7

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

Recent analyses have been performed by Westinghouse to recalculate site boundary and control room doses from a large break LOCA, which is the most limiting LOCA size for dose evaluations. With an assumed recirculation leak rate of 2.75 GPH, it was determined that the site boundary and low population zone thyroid and whole body doses resulting from the postulated large break LOCA are within the 10 CFR 100 dose guidelines of 300 REM thyroid and 25 REM whole body, respectively. The Control Room dose calculations result in a Control Room thyroid, whole body and beta skin dose within the dose limits defined in the NRC Standard Review Plan (SRP) Section 6.4, which are 30 REM thyroid, 5 REM whole body and 30 REM beta skin. These calculated doses are also within the dose values stated in the UFSAR.

Based on the above, it can be concluded that the public's health and safety was assured at all times.

V. CORRECTIVE ACTION:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

- The source of the leakage was isolated from the RWST by freeze seal, permitting exit from ITS LCO 3.0.3.
- The affected pipe and socket welds were cut out and replaced with a new prefabricated spoolpiece, and the freeze seal was removed.

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

- Failure Analysis of the leak in the socket weld has been performed.
- The adjacent original existing welds were cut out and replaced.
- Additional corrective actions are identified in this supplement and are listed below:

1. The bases for ITS LCO 3.6.1, "Containment", has been revised to specify the new leakage limit of 2.75 GPH for integrated leakage outside CNMT from the residual heat removal (RHR), safety injection (SI), and CS systems, and to consider the effect of this leakage on CNMT operabiity.
2. The program limit for leakage outside CNMT, as stated in procedure PT-39, was revised from 2.0 GPH to 2.75 GPH, to reflect these new leakage limits.
3. Procedure A-52.4 was revised to reflect the new program limit: the integrated leakage must be </= 2.75 GPH as determined by PT-39. The conservative requirement to enter ITS LCO 3.0.3 for leakage above this limit was eliminated.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
R.E. Ginna Nuclear Power Plant	05000244	96 --	009 --	02	7 OF 7

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

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The failed component was a 2 inch nominal pipe size Schedule 10 pipe-to-90 degree forged socket elbow weld. The pipe and socket elbow are stainless steel, ASTM A312 Type 304 and A182 F304 (forged) respectively. The socket weld was an original installation weld made before 1970.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events with the same root cause at Ginna Nuclear Power Plant could be identified.

C. SPECIAL COMMENTS:

None