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the southern electric system

February 19, 1997

Docket No. 50-321

HL-5318

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

Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Pressure Boundary Leakage Results in
Condition Prohibited by the Technical Specifications

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(i), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning pressure boundary leakage which resulted in a condition prohibited by the Technical Specifications.

Sincerely,

J. D. Woodard

IFL/eb

Enclosure: LER 50-321/1997-001

cc: *Georgia Power Company*
Mr. H. L. Sumner, Nuclear Plant General Manager
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

IE22

NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000321	1 OF 6

TITLE (4)
Pressure Boundary Leakage Results in Condition Prohibited by the Technical Specifications

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER(S)
01	29	97	97	001	00	02	19	97		05000
										05000

OPERATING MODE (9): 1
POWER LEVEL (10): 010

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)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		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 (include area code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	912 367-7851

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
X	BO	PSP	Q015	Yes					

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-space typewritten lines) (16)

On 1/29/97 at 0835 EST, Unit 1 was in the Run mode at a power level of approximately 256 CMWT (10 percent rated thermal power). Power had been reduced in order to identify and repair the source(s) of leakage into the drywell floor drain sump. At that time, it was determined leakage was coming from the area of a socket weld on a 3/4-inch line connected to the Residual Heat Removal (RHR) system suction line. The location of the leak was such that it met the Technical Specifications definition of pressure boundary leakage, that is, leakage through a nonisolable fault in the reactor coolant system. Limiting Condition for Operation (LCO) 3.4.4 allows no pressure boundary leakage. Therefore, Operations personnel placed the mode switch in the Shutdown position, shutting down the unit as required by LCO 3.4.4, Required Action C.1. By 2125 EST, the leak had been isolated using manual valve 1E11-F067 and personnel had confirmed the leakage had ceased. By 0725 EST on 1/30/97, plant Maintenance personnel had repaired the line. Valve 1E11-F067 was re-opened.

The cause of this event can not be determined conclusively. However, it appears it may have been caused by a combination of a weld anomaly or discontinuity and high cycle fatigue. The existing weld was removed, the line was shortened and re-welded to the RHR suction line, and dye penetrate and magnetic particle tests were performed successfully on the new weld.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 1/29/97, Unit 1 was in the Run mode at a power level of approximately 256 CMWT (10 percent rated thermal power). Power had been reduced in order to identify and repair the source(s) of leakage into the drywell floor drain sump (EIIS Code IJ). Drywell floor drain sump in-leakage, that is, unidentified leakage into the drywell, had increased linearly from approximately 0.1 gpm at the end of August 1996 to approximately 2.4 gpm at the end of January 1997. Although the Unit 1 Technical Specifications limit on unidentified leakage is five gpm, plant management decided, conservatively, to reduce power and identify and repair the sources of the unidentified leakage before the Technical Specifications limit was approached.

After entering the drywell, plant personnel discovered leakage coming from the area of a socket weld on a 3/4-inch line connected to the Residual Heat Removal (RHR, EIIS Code BO) system suction line. The location of the leak was such that it met the Technical Specifications definition of pressure boundary leakage, that is, leakage through a nonisolable fault in the reactor coolant system. Unit 1 Technical Specifications Limiting Condition for Operation (LCO) 3.4.4 allows no pressure boundary leakage.

At 0835 EST on 1/29/97, Operations personnel placed the mode switch in the Shutdown position thereby shutting down the unit and placing it in the Hot Shutdown mode as required by LCO 3.4.4, Required Action C.1. They also initiated action to ensure the unit would be in cold shutdown within the time directed by Required Action C.2. Additionally, a notification of unusual event was declared as required by Plant Hatch's Emergency Plan based upon a nonisolable leak in the reactor coolant pressure boundary. By 2125 EST, the leak had been isolated using manual valve 1E11-F067 (located in the drywell and, hence, inaccessible during normal plant operation) and personnel had confirmed the leakage had ceased. By 0725 EST on 1/30/97, plant Maintenance personnel had repaired the line. Valve 1E11-F067 was re-opened and the notification of unusual event was terminated.

CAUSE OF EVENT

The root cause of this event can not be determined conclusively. The portions of the pipe and weld which might have failed were destroyed in the process of removing the existing weld and line from the socket. Thus, no conclusive determinations regarding the failure mechanisms could be made. However, based on plant history with this type of line installation, engineering experience and judgment, and input from Hatch's Architect/Engineer, it appears possible that the cause of this event was a combination of a weld anomaly or discontinuity and high cycle fatigue. Engineering experience and stress analysis indicate that in this particular installation neither cause alone would have produced the failure which occurred. Therefore, it is concluded that these factors probably combined to produce the failure of the pressure boundary weld.

The particular line which failed in this event had been replaced per Maintenance Work Order 1-96-1054 during the Spring 1996 refueling outage. During that replacement, the total length of the line appears to have been increased by approximately 13 inches. This would have had the effect of increasing the normal vibration-induced stress occurring in the vicinity of the weld. However, Hatch's Architect/Engineer analyzed this piping installation and determined that vibration-induced stress (high-cycle fatigue) alone would not have caused the line to fail in its longer configuration. Hence, some factor which would exacerbate the effects of high cycle fatigue, such as a weld anomaly or discontinuity, is believed to have existed.

The tentative conclusion that a weld anomaly or discontinuity existed is strengthened based on interviews with the individual who performed the removal of the 3/4-inch line from the socket and the Quality Control inspector who visually examined the leak. Based upon their observations, it appears that the failure occurred near the base of the weld, or possibly through the weld, and not on the pipe above the upper portion of the weld as would normally be expected if high cycle fatigue alone were the failure mechanism. Hence, it is concluded that a stress riser also was present in or near the weld.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(i) because a condition existed which was prohibited by the plant's Technical Specifications. Unit 1 Technical Specifications LCO 3.4.4 allows no pressure boundary leakage. The long-term increasing trend in drywell floor drain sump in-leakage and the discovery of a leak at the connection of a 3/4-inch line to the RHR suction line indicated a leak in the pressure boundary had existed for longer than allowed by LCO 3.4.4. Therefore, the plant was in a condition prohibited by the Unit 1 Technical Specifications.

The reactor coolant system includes systems and components that contain or transport the coolant to or from the reactor core. The pressure-retaining components of the reactor coolant system and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. Limits on reactor coolant system operational leakage are required to ensure appropriate action is taken before the integrity of the reactor coolant pressure boundary is impaired. The Technical Specifications specify the types and limits of leakage.

The unidentified leakage flow limit allows time for corrective action before the reactor coolant pressure boundary can be compromised significantly. The five-gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. A critical crack is one large enough to propagate rapidly, ultimately leading to failure of the affected component. Crack behavior from experimental programs shows that leakage rates of hundreds of gallons per minute will precede crack instability (see Unit 1 Final Safety Analysis Report, section 4.10, "Nuclear System Leakage Detection and Leakage Rate Limits," and Unit 1 Technical Specifications Bases B 3.4.4, "RCS Operational Leakage").

In this event, unidentified leakage into the drywell had increased linearly from approximately 0.1 gpm at the end of August 1996 to approximately 2.4 gpm at the end of January 1997. After reducing power to find and repair the sources of the unidentified leakage, personnel discovered a nonisolable pressure boundary leak in the area of a socket weld on a 3/4-inch line connected to the RHR system suction line. Upon identification of the leak as a nonisolable pressure boundary leak, Operations personnel shut down the unit.

At the time the unit was shut down, the unidentified leakage rate was less than one-half of its Technical Specifications-allowed limit of five gpm. The size of the crack was much smaller than the critical crack as evidenced by the low leakage rate. Therefore, the crack was not unstable and would not have resulted in catastrophic failure of the line. However, a worst-case (and unrealistic) instantaneous and complete severing of the 3/4-inch line due to the presence of the crack at this low leakage rate would not result in a significant loss of reactor coolant or present any challenge to core cooling.

Had the pipe severed at the area of the crack, the maximum flow rate through the resulting opening would have been approximately 410 gpm. This is less than 10 percent of the rated capacity of the High Pressure Coolant Injection (EIIS Code BJ) system, which is sized to provide adequate coolant make-up for pipe breaks up to four inches, and about the rated capacity of the Reactor Core Isolation Cooling (EIIS Code BN) system. It should be noted the calculation assumed only liquid

flows out of the resulting opening; in reality, a combination of liquid and vapor would flow from the break area. The actual, two-phase flow rate would be lower, perhaps by as much as a factor of three, than that resulting from liquid only. Consequently, either system would have been capable of indefinitely maintaining normal reactor water level. Additionally, a leak of a few hundred gpm easily would have been accommodated by the feedwater system (EIIS Code SJ) which has a flow rate capacity margin at rated conditions of at least 10 percent (over 2000 gpm). Therefore, any one of three diverse and independent high pressure injection systems could have provided sufficient make-up flow to maintain water level well above the top of the active fuel.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety. This analysis is applicable to all operating conditions under which the crack might have propagated to line failure.

CORRECTIVE ACTIONS

The existing socket-to-pipe weld was removed, the line was shortened by approximately seven inches and re-welded to the RHR suction line, and dye penetrate and magnetic particle tests were performed successfully on the new weld per Maintenance Work Order 1-97-173. Following repair of the line and other, minor leaks (e.g., valve packing leaks), unidentified leakage in the drywell decreased to less than 0.1 gpm.

In order to reduce any possible stress risers at the pipe-to-weld interface, a larger weld (1/4-inch by 1/2-inch) was used in re-attaching the line and the weld was contoured to decrease or eliminate undercut. The performance of two weld surface tests, the use of a larger weld, and the contouring of the weld to eliminate undercut exceeded American Society of Mechanical Engineers Code requirements applicable to this 3/4-inch line.

The line's as-left configuration was analyzed for vibration-induced stress by Hatch's Architect/Engineer and was determined to be acceptable.

An investigation was conducted for the existence of circumstances similar to the ones which may have led to this event, specifically, small bore, Class 1, cantilevered piping welded during the Spring 1996 Unit 1 outage. No similar conditions were found to exist; therefore, it may be concluded this event was isolated.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

Failed Component Information:

Master Parts List Number: N/A
Manufacturer: Quanex
Model Number: N/A
Manufacturer Code: Q015
Type (pipe): 3/4-inch, schedule 160, SA-106, grade B, Class 1, seamless carbon steel

EIIS System Code: BO
Reportable to NPRDS: Yes
Root Cause Code: X
EIIS Component Code: PSP

The weld, 1/4-inch by 1/4-inch, nominal, was made using 3/32-inch, ER705-2 weld rods.

There have been no previous similar events in the last two years in which a nonisolable fault existed in the reactor coolant system.