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Georgia Power
the southern electric system

Docket No. 50-366 November 8, 1995 HL-5061

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Excessive Leakage Identified on
Secondary Containment Bypass Valves

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(ii), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning the Local Leak Rate Testing results of two Secondary Containment bypass valves.

Sincerely,

J. T. Beckham, Jr.

JKB/eb

Enclosure: LER 50-366/1995-004

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

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

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

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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 2	05000366	1 OF 5

TITLE (4)
Excessive Leakage Identified on Secondary Containment Bypass Valves

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)
10	09	95	95	004	00	11	08	95		05000
										05000

OPERATING MODE (9): 5
POWER LEVEL (10): 000

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)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	X 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	IJ	ISV	P032	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 10/9/95 at 0855 EDT, Unit 2 was in the Refuel mode with no fuel in the vessel and the reactor cavity flooded for refueling operations. At that time, plant engineers and technicians were performing Local Leak Rate Testing (LLRT) on Primary Containment Isolation Valves (PCIVs) when it was discovered that two "Secondary Containment bypass valves" had failed their associated LLRT. Both valves are located in the same penetration involving the drywell equipment and floor drain sumps. The valves are classified as "Secondary Containment bypass valves" as the drain piping is routed to the Radwaste Building which is not inside the Secondary Containment and, consequently, not served by the Standby Gas Treatment System.

The primary cause of the failures of these two valves appears to be the degradation of the valves over a period of time to the orientation of the valves in a non-vertical position. This orientation results in a number of problems including accelerated corrosion, increased wear, uneven seating of the disc, and binding. Corrective actions for this event included replacing one valve with a new valve and restoring the other valve to a condition with essentially zero leakage. In addition, other options are being considered, such as moving the valves to a different location on this line so they can be oriented vertically, or changing the valve or operator. A Request for Engineering Assistance has been sent to the Architect/Engineer to evaluate these options and determine which, if any, is the best course of action to prevent recurrence.

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 10/9/95 at 0855 EDT, Unit 2 was in the Refuel mode with no fuel in the vessel and the reactor cavity flooded for refueling operations. At that time, engineers and technicians were performing Local Leak Rate Testing (LLRT) on valves 2G11-F003 and 2G11-F004 per Unit 2 Technical Specifications surveillance requirement (SR) 3.6.1.3.10. This SR addresses the leakage restrictions for "Secondary Containment bypass valves." Secondary Containment bypass valves are valves located in pipes which penetrate the Primary Containment and are open to areas outside the Secondary Containment, i.e., areas not served by the Standby Gas Treatment System (SBGT, EIIS Code BH). The subject valves, 2G11-F003 and 2G11-F004, are located in the same Primary Containment penetration serving the drywell equipment and floor drain sumps which drain to the Radwaste Building (EIIS Code NE), an area not served by the SBGT system. The acceptance criterion for leakage of these kinds of valves is less than 0.9 percent of the total allowable leakage from the Primary Containment (designated L_a). For Plant Hatch, this would be a maximum allowable leakage of 544 actual cubic centimeters per minute (ACCM). The actual measured leakage for 2G11-F003 was 17,858 ACCM and for 2G11-F004 it was greater than 30,000 ACCM (off scale of the measuring device being used).

CAUSE OF EVENT

The primary cause of the failure of these two valves appears to be the degradation of the valves over a period of time due to the orientation of these valves in a non-vertical position. Both of these valves are air cylinder-operated and are located in an area where there is little clearance for the valve operators. Consequently, both valves are oriented with the air cylinder and valve bonnet positioned at about 30 degrees below horizontal. This results in water standing in the valve bonnet, which accelerates normal corrosion rates. In addition, the orientation of the valves results in the disc riding on the guide ribs during stroking which increases the frictional forces between the disc and ribs and increases the wear on surfaces affected by accelerated corrosion. In this case, wear on these surfaces resulted in a slight misalignment between the disc and seat such that the valve did not seal properly in the closed position.

In addition to problems related to valve orientation, preliminary engineering analysis on this valve suggests the operator may have been sized with too little stem thrust margin over the frictional forces between the packing and the stem. Valve diagnostic testing showed that the natural decrease in actuator spring constant expected over time could have resulted in the stem thrust being insufficient to ensure proper seating of the valve.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73 (a)(2)(ii) because an event occurred which resulted in the degradation of one of the plant's principal safety barriers. Specifically, two Secondary Containment bypass valves failed to pass Local Leak Rate Testing requirements.

The function of the Primary Containment is to isolate and contain fission products released from the Reactor Primary System following a design basis accident (DBA) and to confine the postulated release of radioactive material. The Primary Containment consists of a steel vessel which surrounds the Reactor Primary System and provides a barrier against the uncontrolled release of radioactive material to the environment. Some leakage from the Primary Containment is assumed to occur, although the majority of the leakage is assumed to be released into the Secondary Containment. The total allowable leakage rate for the Primary Containment is designated L_a and is equal to 1.2 percent by weight of the contained air volume per day. For Plant Hatch Unit 2, this equates to a total allowable leakage of 36,244 ACCM, most of which is assumed to occur within the Secondary Containment where it will be treated by the Standby Gas Treatment System before being released at an elevated point through the Main Stack (EIIS Code VL). However, some small amount of leakage is assumed to occur outside Secondary Containment where it is released without being treated by the SBGT system. Valves located in Primary Containment penetrations whose pipes lead outside the Secondary Containment are potential sources of such untreated leakage, so these valves are termed "Secondary Containment bypass valves." Since the atmospheres in such areas would not be treated by the SBGT system, the allowable leakage through these valves is specifically addressed by the Technical Specifications, and is limited to a total of 544 ACCM. The leakage rates measured in this event were greater than this amount.

The allowable leakage for Secondary Containment bypass valves was established using conservative licensing basis evaluation methods in accordance with NRC Regulatory Guide 1.3. These methods conservatively assume that the postulated accident results in fuel damage with 100 percent of the core noble gas activity and 50 percent of the iodine activity released. Consequently, the actual measured leakage of the valves identified in this report would likely have resulted in exceeding the values set forth in 10 CFR 100 during a postulated design basis accident that assumes fuel damage per NRC Regulatory Guide 1.3.

The Final Safety Analysis Report (FSAR) for Plant Hatch Unit 2 designates the Design Basis Accident (DBA) as the break of a Reactor Recirculation System (EIIS Code AD) pipe which results in the rapid depressurization of the reactor vessel to the Primary Containment. However, the FSAR analysis shows that, for such an accident, resulting peak fuel cladding temperatures would be less than those required to produce damage to the fuel. The plant-specific SAFER/GESTR analysis for this accident scenario shows that no damage to the fuel cladding would occur even if additional failures are postulated, such as failures of certain power supplies and certain low pressure emergency core cooling systems. Therefore, by this analysis, the only radioactive materials present in the released coolant would be those already present due to normal operation and the small additional amount of contaminated or activated crud released from vessel internals and primary system piping during the initial stages of the transient. If it is conservatively assumed that all reactor coolant released into the Primary Containment drains through valves 2G11-F003 and 2G11-F004 into the Unit 2 Radwaste Building where is it released to the environment (neglecting plateout), this would represent an unfiltered, ground level release of all the radioactive materials present in the coolant at the time of the accident. Under these extremely conservative assumptions, calculations performed by the Architect/Engineer show that the dose rate at the site boundary would still be very much less than the limits prescribed by 10 CFR 100.

Based on this analysis, it is concluded that this event did not result in any adverse impact on nuclear safety. This analysis applies to all operating conditions.

CORRECTIVE ACTIONS

1. Valve 2G11-F003 was refurbished and valve 2G11-F004 was replaced. Prior to replacing 2G11-F004, both valves were the original valves installed during plant construction and have been in service for approximately 15 years. Although the orientation contributed to an LLRT failure, the failure mechanism involves a time frame longer than one or two operating cycles to produce degradation where the valves do not perform satisfactorily. Therefore, replacement of the 2G11-F004 valve is considered to be a satisfactory solution for at least the next operating cycle. Furthermore, valve 2G11-F003 has failed the associated LLRT only twice since 1980. In both of those cases, the valve repair was successful and the refurbished valve passed the next LLRT in its as-found condition during the following refueling outage. Thus, plant operating and maintenance history with these valves indicates that they may be operated in their present configuration for at least one operating cycle with reasonable assurance the associated Primary Containment penetration will remain within its acceptance criteria.

In addition, both the new valve and the refurbished valve were installed with an improved composite packing designed to reduce packing friction and increase stem thrust margin to an acceptable level. The leakage characteristics for both valves were retested and the valves were left with essentially zero leakage. These actions are complete.

2. A Request for Engineering Assistance has been sent to the Architect/Engineer for Unit 2 for the purpose of identifying methods to prevent recurrence of the failures of these valves. Specific avenues of corrective action to be explored involve design changes to the valve orientation, valve design, valve location, and actuator design. The evaluation will be completed prior to the 1997 Unit 2 refueling outage.

ADDITIONAL INFORMATION

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

2. Failed Component Information:

   Master Parts List Number: 2G11-F003/F004
   Model Number: NP8316A54E
   Manufacturer Code: P032
   Reportable to NPRDS: Yes
   EIIS Component Code: ISV

   Manufacturer: Pacific Valves Inc.
   Type: Air Operated Gate Valve
   EIIS System Code: IJ
   Root Cause Code: X

3. Previous Similar Events: No events have been reported in the past two years in which Secondary Containment bypass valves failed LLRT, in part because previous Technical Specifications did not require leak rate tests to use maximum pathway leakage in the acceptance criteria as the current Technical Specifications do. Also, there have been no events in the past two years in which both Secondary Containment bypass valves in the same penetration failed LLRT.