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Georgia Power
the southern electric system

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

PLANT HATCH - UNIT 1
NRC DOCKETS 50-321
OPERATING LICENSES DPR-57
LICENSEE EVENT REPORT
DESIGN DEFICIENCY COULD AFFECT
MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM

Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(v), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning a condition that could have prevented an ESF from fully performing its safety function. This event occurred at Plant Hatch - Units 1 and 2.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

SRB/CT/cr

Enclosure: LER 50-321/1991-009

cc: (See next page.)

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Georgia Power

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. L. D. Wert, Senior Resident Inspector - Hatch

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NRC Form 366 (6-89) U.S. NUCLEAR REGULATORY COMMISSION APPROVED OMB NO. 3150-0104 EXPIRES: 4/30/92

LICENSEE EVENT REPORT (LER)

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TITLE (4)
DESIGN DEFICIENCY COULD AFFECT MAIN CONTROL ROOM ENVIRONMENTAL CONTROL SYSTEM

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
									PLANT HATCH, UNIT 2	0 5 0 0 0 3 6 6
0 7	1 2	9 1	9 1	0 0 9	0 0	0 8	0 5	9 1		0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)

OPERATING MODE (9): 1
POWER LEVEL: 1 0 0

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[X] 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)
[ ] 20.405(a)(1)(iii)	[ ] 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 AREA CODE	
STEVEN B. TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912	367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

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

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (16)

On 7/12/91, at approximately 1205 CDT, Units 1 and 2 were in the Run mode at 2436 CMWT (100 percent of rated thermal power). At that time, nonlicensed personnel determined that the Main Control Room Environmental Control (MCREC) system did not comply with the single failure design criterion as required by the plant's Final Safety Analysis Report. Specifically, the air conditioning subsystem of the MCREC system could not sustain a single failure to the system's class 1E power supply and still maintain the Main Control Room temperature within the Technical Specifications limits. Consequently, it was determined that only one MCREC system was operable contrary to the plant's Technical Specifications which require that two independent systems be operable. A limiting condition of operation (LCO) was entered per the Technical Specifications. On 7/16/91, a design change and a procedure revision were completed bringing the system into compliance with the single failure design criterion. The LCO was subsequently terminated.

The cause of the event was less than adequate design of the system.

Corrective actions include implementing design changes to the system and revising a procedure to bring the system into compliance with the single failure design criterion. Also, a design review of the system is being performed to determine if other problems exist in relation to the single failure design criterion.

PLANT AND SYSTEM IDENTIFICATION

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

DESCRIPTION OF EVENT

On 7/12/91, at approximately 1205 CDT, Units 1 and 2 were in the Run mode at 2436 CMWT (100 percent of rated thermal power). At that time, nonlicensed Nuclear Safety and Compliance personnel determined that the air conditioning subsystem of the Main Control Room Environmental Control (MCREC, EIIS Code VI) system did not meet the single failure criterion specified in the plant's Final Safety Analysis Report (FSAR). Specifically, a single failure resulting in the loss of one division of the Class 1E electrical power supply to the MCREC system would result in the air conditioning subsystem operating at 50 percent of its design capacity and, thus, would be unable to maintain the Main Control Room (EIIS Code NA) at or below 105 degrees Fahrenheit as required by Unit 2 Technical Specifications, section 4.7.2.a. With the single failure criterion not met, only one as opposed to two independent MCREC trains could be assumed to be operable, contrary to the requirements of Unit 1 Technical Specifications, section 3.12.A.1.a and Unit 2 Technical Specifications, section 3.7.2.a. (The MCREC system is shared by both units.) Deficiency Card 1-91-3110 was written to document the condition and track corrective actions. Licensed personnel were notified and Limiting Conditions for Operation (LCOs) 1-91-364 (for Unit 1) and 2-91-519 (for Unit 2) were initiated per the respective unit's Technical Specifications.

The MCREC air conditioning subsystem consists of three 50 percent capacity trains. Each train includes an air handling unit (1Z41-B003A, B, and C), a refrigeration unit (1Z41-B008A, B, and C), and support equipment. Two trains, trains 'A' and 'C', are normally in operation providing 100 percent cooling capacity and train 'B' is normally in standby. The standby train is designed to start automatically on a low flow condition occurring in either of trains 'A' or 'C'.

The MCREC air conditioning subsystem receives power from two independent and redundant Class 1E essential buses (EIIS Code EB). Division I bus 1R24-S002 supplies power to Train 'A'. Division II bus 1R24-S003 supplies power to Train 'B' - the standby train. Swing bus 1R24-S029 supplies power to Train 'C'. The swing bus can be configured to receive power from Division II bus 1R24-S003 (the normal supply) or from Division I bus 1R24-S002 (the alternate supply).

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TEXT

The normal system line-up has the 'A' and 'C' trains in operation with train 'C' being powered from Division II bus 1R24-S003 via swing bus 1R24-S029. Train 'B' was designed to provide backup cooling in the event that either train 'A' or 'C' became inoperative. If train 'A' became inoperative, then trains 'C' and 'B' would be powered from Division II bus 1R24-S003. If Division II bus 1R24-S003 failed, bus 1R24-S029 could be transferred to the Division I bus 1R24-S002 restoring power to train 'C'. In this case, trains 'A' and 'C' would be operating and both powered from Division I bus 1R24-S002.

In 1989, plant personnel requested the Architect Engineer to specify which power supply for 1R24-S029 was the preferred normal supply and which was the alternate supply. The Architect Engineer, in responding to the request, evaluated the loading of the buses and determined that operating the 'A' and 'C' trains concurrently and powered from the same bus, 1R24-S002, or operating 'B' and 'C' trains concurrently and powered from the same bus, 1R24-S003, would result in overloading the feeder cables to the applicable bus. To address this problem, the Architect Engineer recommended that train 'C' be aligned to the divisional bus that was not supplying power to the other operating train (i.e., Division II bus 1R24-S003 if train 'A' were in operation or Division I bus 1R24-S002 if train 'B' were in operation). Accordingly, procedure 34SO-Z41-001-1S, "Control Room Ventilation System," was revised to incorporate the recommendation.

On 7/12/91, Nuclear Safety and Compliance personnel had been reviewing the adequacy of the procedural instructions in 34SO-Z41-001-1S for transferring the power supply for train 'C' when they determined that the inability to load two of the system trains simultaneously on one Class 1E divisional bus presented a single failure concern. In particular, if each bus could only power one train, then loss of either bus would result in only one train being operable, which is insufficient for cooling the Main Control Room. Personnel also noted in the review that the power supply configuration for the controls of train 'C' also presented a single failure concern. The 'C' train controls have a dedicated power supply, Class 1E Division II bus 1R24-S003, whereas the 'C' train electrical components are powered from the swing bus 1R24-S029. This configuration would result in a loss of control power to the 'C' train in the event that the Division II bus were inoperable. Personnel subsequently wrote a deficiency card on the two deficient conditions and notified licensed personnel.

Design Change Request 1H91-130 was developed and implemented to resolve the power supply problem for the train 'C' controls. The power distribution system has been reconfigured so that upon a loss of power to the train 'C' controls from the normal supply, Division II bus 1R24-S003, a transfer can be made to the Division I bus 1R24-S002. Regarding the potential overload problem, an evaluation of the loads on buses 1R24-S002 and 1R24-S003 showed that several specific loads can be disconnected from the buses so that two trains can be powered from one bus without creating an overload condition. Procedure 34SO-Z41-001-1S was revised to require disconnecting selected loads should two trains have to be powered from the same bus. Each buses' feeder cables are sized to handle the resulting loads.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT

The described changes [illegible] completed by 7/16/91. LCO's 1-91-364 and 2-91-519 were subsequently ter[illegible]d at approximately 1630 CDT, on 7/16/91.

CAUSE OF EVENT

The cause of this event is less than adequate design. The architect engineer did not sufficiently evaluate the power supply scheme to the MCREC air handling units/compressors to ensure that the required single failure design criterion was met.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(v) because the power supply design for the MCREC system was such that a single failure could prevent the fulfillment of its safety function.

The purpose of the MCREC air conditioning subsystem is to maintain the Main Control Room temperature within acceptable limits during normal plant operations and following an accident to ensure Main Control Room equipment reliability and Main Control Room habitability.

A conservative analysis was performed to determine the impact that operating the air conditioning system at 50 percent capacity would have on the Main Control Room temperature. Some of the conservative assumptions were as follows. The temperature of the ultimate heat sink for the MCREC system, the Plant Service Water System (EIIS Code BS), was assumed to be at the maximum design limit of 95 degrees Fahrenheit. The Turbine Building (EIIS Code NM) which houses the Main Control Room was assumed to be at 110 degrees Fahrenheit, the maximum temperature expected during normal operation. Also, the outside ambient air temperature was assumed to be 95 degrees Fahrenheit. Based on the analysis, should the MCREC system be reduced to 50% capacity, the Main Control Room could potentially reach a temperature of approximately 120 degrees Fahrenheit in 40 minutes. At this temperature the Main Control Room would be considered uninhabitable and the Main Control Room instrumentation reliability questionable.

The MCREC system provides support for systems designed to perform a safety function in that it affords habitability of the Main Control Room during normal plant operation and following a design basis accident. In an assumed worst case scenario, the single failure addressed in this report could occur coincident with a design basis accident such as a LOCA or a Main Steam Line break. In such

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TEXT

an event, safety related systems would function automatically to shutdown the reactor and restore it to stable conditions within minutes following the initiating event. Consequently, ample time would be available to ensure that the reactor is stable before the Main Control Room temperature reaches 120 degrees Fahrenheit necessitating evacuation of the Main Control Room. Prior to the Main Control Room becoming uninhabitable, operation of each unit could be transferred to the Remote Shutdown system. The Remote Shutdown system has the capability for prompt hot shutdown of the reactor, including necessary instrumentation and control to maintain the unit in a safe condition during hot shutdown, and subsequent cold shutdown of the reactor through use of administrative procedures.

It is postulated that within 24 hours, the MCREC air conditioning subsystem could be restored to 100 percent operating capacity. Following cooldown of the air space and testing of instrumentation, operation of the plant could then be transferred back to the Main Control Room.

Based on the above information, this event had no adverse affect on nuclear safety. This analysis applies to all operating conditions.

CORRECTIVE ACTION

DCR 1H91-130 was implemented to provide an alternate power supply for the train 'C' controls in the event that the Division II power supply is inoperative. The DCR was completed on 7/16/91.

Procedure 34SO-Z41-001-1S has been revised to provide instructions for disconnecting specific loads from buses 1R24-S002, 1R24-S003, or 1R24-S029 to allow the operation of two air conditioning trains powered from the same bus without causing an overload condition. This is a temporary corrective action. The feeder cables to buses 1R24-S002 and 1R24-S003 will be replaced with larger capacity cables during the next Unit 1 Refueling Outage currently scheduled to begin 9/18/91. At that time selective load shedding of the buses will no longer be required and the procedural instructions will be deleted.

As mentioned in the "Additional Information" section of this report, three previous similar events have been identified in which the MCREC system design was found to deviate from the single failure design criterion. In each case, the design was corrected to bring the system into compliance with the design requirement. These examples may be indicative of a generic problem with the design of the system. Consequently, a design review of the system will be performed to evaluate it against the single failure design criterion. This review will be completed by 12/31/91.

ADDITIONAL INFORMATION

No systems other than the MCREC system were affected by this event.

TEXT

Three previous similar events have been identified in which the MCREC system was determined not to be in compliance with the single failure design criterion. These events were reported in LERs 50-321/87-04, Revision 1, dated 8/8/88 and 50-321/88-11, dated 6/8/88. The first event, reported in LER 50-321/87-04, Revision 1, involved a single fuse failure preventing the MCREC system from fully entering the isolation mode. The second event, also reported in LER 50-321/87-04, Revision 1, involved a failure of one chlorine gas monitor preventing the MCREC system from fully entering the pressurization mode. The third event involved the use of non-seismic area radiation monitors in the MCREC system pressurization mode actuation logic system. Failure of the monitors during a seismic event could have possibly grounded the actuation logic circuits rendering them inoperable and preventing the system from entering the pressurization mode.

Corrective actions for these events included design changes in each case to bring the system into compliance with the single failure design criterion. These corrective actions would not have prevented this event since the portion of the system involved in this event was unique to this event.