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Georgia Power
the southern electric system

December 14, 1995

Docket No. 50-366 HL-5086

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Remote Shutdown Panel Found Degraded
Due to Inadequate Testing and Design

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 degraded remote shutdown panel functions.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

OCV/eb

Enclosure: LER 50-366/1995-009

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 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Plant Edwin I. Hatch, Unit 2	05000366	1 OF 8

TITLE (4)
Remote Shutdown Panel Found Degraded Due to Inadequate Testing and Design

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 NAMES	DOCKET NUMBER(S)
11	14	95	95	009	00	12	14	95	Plant Hatch, Unit 1	05000321

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

OPERATING MODE (9): 4
POWER LEVEL (10): 000

[ ] 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 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)	[ ] 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, Nuclear Safety & 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
E	JG	NONE	G080	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

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

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

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

On 11/14/95, Unit 2 was in Cold Shutdown, preparing for startup from the twelfth refueling outage. At that time, an assessment of previously identified deficiencies on the Unit 2 Remote Shutdown Panel System (RSDP) showed that several of the engineered safety features of the system would have been unable to function as designed. Specifically, Recirculation System loop "B" suction valve and several components of the Residual Heat Removal System were incapable of being operated from the RSDP. The ability to operate these components from the Main Control Room was unaffected by the noted deficiencies. The deficient conditions had been identified between 10/20/95 and 11/11/95 during performance of new surveillance testing required under the Improved Technical Specifications (ITS). The unit was in a refueling outage at the time the conditions were discovered. Since the RSDP is not required to be operable with the unit shut down, no Technical Specification actions were required to be entered. The conditions were corrected prior to the unit starting up from the refueling outage on 11/18/95. The causes of the conditions were failure to perform periodic testing of the RSDP and, in one case, design error. As a lead plant in the ITS program, Georgia Power Company had implemented the Improved Technical Specifications at Plant Hatch on 7/13/95 which included RSDP surveillance requirements. Corrective actions included correcting each of the conditions identified, completion of the ITS surveillances, training, Unit 1 RSDP testing, and continuing performance of ITS-RSDP surveillances.

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 11/14/95, Unit 2 was in Cold Shutdown, preparing for startup from the twelfth refueling outage. At that time, an assessment of previously identified deficiencies on the Unit 2 Remote Shutdown Panel System (RSDP, EIIS Code JG) showed that several of the system engineered safety features would have been incapable of performing their intended function as designed. The deficient conditions had been identified between 10/20/95 and 11/11/95 during performance of RSDP system testing. The tests were performed to comply with new surveillance requirements recently issued under the Improved Technical Specifications.

The RSDP provides the capability for control outside of the Main Control Room (MCR, EIIS Code NA) of systems needed to shutdown the reactor and maintain it in a shutdown condition in the event that the MCR becomes uninhabitable. Prior to 7/13/95, the Unit 2 Technical Specifications did not require testing of the RSDP transfer circuits (designed to transfer control of associated components from the MCR to the RSDP) nor the RSDP component control circuits; and, no periodic testing of these circuits had been performed. Plant E. I. Hatch was a lead plant in the implementation of the Improved Technical Specifications (ITS) and on 7/13/95, implemented ITS. ITS included requirements for performing surveillance tests on the RSDP transfer and control circuits.

The ITS surveillances required that circuits for a number of the components which can be operated from the RSDP be tested once per 18 months. At the time that ITS was implemented, Unit 2 was in operation. The surveillance procedures developed to implement the surveillance requirements included cycling components from the RSDP. Cycling some of the components from the RSDP posed a risk to the continued operation of the unit and rendered inoperable associated safety systems during testing. Therefore, the Nuclear Regulatory Commission (NRC) allowed postponement of the RSDP surveillances until the next refueling outage, scheduled to begin on 9/23/95. The RSDP is not required to be operable when the reactor is shut down. Therefore, the surveillances could be performed during the outage and still meet the ITS limiting conditions for operation.

Five conditions were discovered during the surveillance testing between 10/20/95 and 11/11/95 which rendered safety systems incapable of performing their intended safety functions when operated from the RSDP. It is noted that none of these conditions affected the ability of the systems to perform their intended function when operated from the MCR. The conditions are as follows:

Residual Heat Removal System (RHR, EIIS Code BO) Discharge Valve (2E11-F017B): During the surveillance, this valve could not be opened or closed from the RSDP. During implementation of Design Change Request (DCR) 87-150 in 1992, a design error was implemented which resulted in this condition. Specifically, a control circuit wire in the valve motor control center (MCC) was landed per the design on a terminal that prevented the control circuit from receiving power when aligned for operation from the RSDP. This valve is normally open, therefore its low pressure coolant injection (LPCI) safety function could have been achieved without operating the valve. This condition did not affect operation of the valve from the MCR.

RHR Heat Exchanger Outlet Valve (2E11-F003B): During this surveillance, this valve could not be opened from the RSDP. Either bent wire lugs or dirty transfer switch contacts prevented the successful transfer of valve control from the MCR to the RSDP. (The transfer switch transfers control of the component from the MCR to the RSDP.) The valve is normally open, which is its safe position. If closed, however, the valve could be manually opened. This condition did not affect operation of the valve from the MCR.

RHR-Shutdown Cooling Suction Valve (2E11-F006A): During surveillance testing, this valve could not be opened from the RSDP. A limit switch contact arm was found out of adjustment The valve is normally closed, which is its safe position. This condition did not affect operation of the valve from the MCR.

RHR-Shutdown Cooling Suction Valve (2E11-F006B): During the surveillance, this valve could not be opened from the RSDP. This valve is interlocked in the opening direction with the RHR Full Flow Test line valve (2E11-F024B) such that if the valve is open, valve 2E11-F006B cannot be opened. The purpose of the interlock is to prevent draining the reactor vessel to the Suppression Pool. It appears that valve 2E11-F024B, on the day of the surveillance, was not fully seated resulting in the interlock limit switch not being closed. This prevented opening of the 2E11-F006B valve from the RSDP. If necessary, the valve could have been operated manually to achieve its safety function. Sufficient time exists, 4 hours, before suppression pool or shutdown cooling is needed. This condition did not affect operation of the valve from the MCR.

Recirculation System (EIIS Code AD) Suction Valve (2B31-F023B): During the surveillance, this valve could not be opened from the RSDP. A loose connection was found in the control power supply circuit which would have prevented the valve from being opened or closed from the RSDP. Safe shutdown could have been achieved with this valve in the open position. This condition had no affect on operation of the valve from the MCR.

By 11/18/95, prior to startup from the refueling outage and prior to the RSDP being required to be operable, the noted conditions were repaired; surveillance testing was satisfactorily completed on each of the components, and the Unit 2 RSDP was returned to operable status.

CAUSE OF EVENT

The causes of this event were failure to perform periodic testing of the RSDP transfer and control circuits and, in one case, inadequate DCR design and functional testing. Additionally, routine operational activities did not involve the RSDP; thus, no previous failures had been identified with the system. As a consequence, periodic reliability testing of the circuits had not been performed. Instrumentation, however, was being calibrated and periodically functionally tested.

In regard to the RHR discharge valve (2E11-F017B), a design error had been implemented in 1992 that involved modifying wiring in the valve MCC. The design specified that a particular control wire be terminated on a specific terminal. This control wire, even though not located in the RSDP, was supposed to have provided power to the valve control logic when valve control was at the RSDP. However, with the wire terminated per the design, the RSDP control logic would not have power when control was transferred to the RSDP. This design error had no effect on operation of the valve from the MCR. The DCR functional test involved cycling the valve from the MCR only and, therefore, did not reveal the condition.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required pursuant to 10 CFR 50.73(a)(2)(v) because a condition existed that could have prevented fulfillment of a safety function from the RSDP of systems designed to remove residual heat from the reactor. Specifically, several of the RHR system components could not be operated as designed from the RSDP without having to take actions beyond those assumed in the event of MCR uninhabitability. It is noted that the capability to operate the affected components from the Main Control Room was not impacted by these conditions.

10 CFR 50, Appendix A, General Design Criterion (GDC) 19 requires in part that capability be provided for shutting down the reactor and maintaining it in a shutdown condition in the event that the MCR becomes uninhabitable for any reason. The RSDP was designed to provide such capability. One of the systems designed with operational capability from the RSDP is the RHR System. The RHR System is designed to be operated from the RSDP in the Shutdown Cooling (SDC) mode and the Suppression Pool Cooling (SPC) mode in meeting the GDC 19 requirement. In the SDC mode of operation, coolant is pumped from the reactor vessel, circulated through a heat exchanger cooled by RHR Service Water (EIIS Code BS), and discharged into the reactor vessel subcore region via the Recirculation system piping and jet pumps. The SPC mode of RHR provides a means for cooling the Suppression Pool. The purpose for the Suppression Pool with regard to GDC 19 is to provide the capability of dissipating the energy of the steam vented from the reactor vessel through the Main Steam Safety Relief Valves (SRVs, EIIS Code SB) during depressurization and cooldown of the reactor vessel. In the SPC mode, water is pumped from the Suppression Pool through a heat exchanger and then back to the Suppression Pool.

10 CFR 50, Appendix R requires that alternative shutdown capabilities be specified for each fire area containing structures, systems and components important to safety. The Remote Shutdown Panel serves as part of the alternate path to the Appendix R safe shutdown path for fires in the main control room, cable spreading room and computer room. In meeting this requirement, operation of RHR in the LPCI mode from the RSDP is assumed. The LPCI mode provides a means to flood the core region following depressurization of the reactor vessel in order to maintain adequate cooling to the core.

The RHR discharge valve (2E11-F017B) provides injection and throttle capability for the LPCI and SDC modes of RHR. This valve is normally open and can be assumed to remain open for meeting shutdown requirements. Since the valve does not have to be cycled for the LPCI function, the inability to open the valve from the RSDP is inconsequential. Also, the valve can be manually throttled as needed when placing SDC in service by manipulating the valve operator handwheel, providing for a controlled reactor vessel cooldown.

Regarding the RHR discharge valve and Appendix R, the most severe analyzed event for a control room fire at Plant Hatch is the spurious operation of an SRV followed by automatic ADS initiation. This event is highly unlikely but is conservatively considered in the Hatch safety shutdown Appendix R analysis. In this situation, LPCI is needed to maintain adequate core cooling. The analysis assumes that the operator initiates LPCI at the RSDP to replenish coolant inventory following the ADS initiation. The 2E11-F017B is normally open and remains open during LPCI operation from the RSDP, therefore, no operation of the valve was necessary and LPCI could have been initiated.

Later in this worst case Appendix R event, SDC is assumed to be placed in service within 4 hours. The safe shutdown procedure directs the operator to close the 2E11-F017B valve from the RSDP then throttle it open. Due to the condition found in this event, the valve could not have been operated from the RSDP. The operator would have to manually close the valve, then throttle the valve open. However, ample time exists to place SDC in service.

It is noteworthy that the Hatch fire procedure requires opening breakers to prevent the ADS initiation, thus making the event even less credible.

The Recirculation system suction valve (2B31-F023B) is normally open and is designed to be closed when RHR is in operation in the SDC mode (and when the Recirculation System is not in operation) and in the LPCI mode. With this valve closed, SDC/LPCI effluent is directed to the reactor core via the Recirculation System discharge piping and the Recirculation System jet pumps, providing adequate cooling for the core. In this event, this valve could not be closed from the RSDP. Most likely, the operator upon determining that the valve would not close would initiate troubleshooting of the problem. If LPCI injection were required, he would proceed to lineup LPCI even with the Recirculation suction valve open. The loose connection would most likely be identified and repaired in a timely manner. The valve would then be closed and, in the case of SDC, the RHR system would be aligned to the SDC mode. Nonetheless, an analysis performed by GE shows that the core can be adequately cooled with the suction valve in the open position provided the Reactor Coolant Pressure Boundary remains intact. With the valve in the open position, the SDC/LPCI effluent would be diverted to the annulus region of the reactor vessel. A path of forced circulation through the core would not be established; however, the core would be maintained in a flooded condition (in LPCI mode) and natural circulation would transfer the cooler water from the annulus region to the core region (in SDC mode). In both cases, the reactor core would be adequately cooled.

The RHR Heat Exchanger Outlet valve (2E11-F003B) is normally open and remains open during the RHR-LPCI mode of operation. Consequently, the inability to open the valve from the RSDP is inconsequential to the LPCI mode. In the SDC and SPC modes of operation, the valve is closed prior to placing the system in these modes in order to prevent water hammer on the heat exchanger internals, maintaining component reliability over the long term. After the RHR pump is started, the valve is opened, routing fluid through the heat exchanger. In this event, the valve could not be opened from the RSDP. If reactor vessel conditions permit, the valve can be closed and then opened with the valve operator, taking approximately 30 minutes. If not, the system can be started with the valve in the open position without adversely affecting the system function. Consequently, inability to open the valve at the RSDP would have no adverse impact on the ability to remove decay heat from the reactor core or to cool the Suppression pool.

RHR-SDC Suction valve 2E11-F006B provides suction source for RHR pump 2E11-C002B, the only RHR pump instrumented for operation from the RSDP. This valve is normally closed and must be opened for the SDC mode of RHR operation. As stated previously, ample time exists for placing SDC in service. Therefore, the valve could be opened manually without adverse consequences. Consequently, the inability to open valve 2E11-F006B from the RSDP had no adverse impact on the SDC mode of operation.

RHR-SDC suction valve 2E11-F006A is instrumented from the RSDP to provide the ability to close the valve prior to placing RHR in the SDC mode of operation. This valve is normally closed. However, if it were open when SDC was placed in service, the potential existed for setting up a drain path from the reactor vessel to the Suppression Chamber via the SDC isolation valves 2E11-F008 and F009, which are opened when placing SDC in service, and the RHR "A" loop minimum flow valve, which is normally open. As such, the condition could be repaired or the valve manually closed prior to placing RHR in the SDC mode.

Based on the above information, it was concluded that this event had no adverse impact on nuclear safety. This assessment applies to all operating conditions.

CORRECTIVE ACTIONS

The deficient conditions were repaired/corrected, including the design error introduced under DCR 87-150.

The ITS-required testing was satisfactorily completed and the RSDP controls returned to operable status prior to reactor startup from the refueling outage.

The surveillance testing on the transfer and control circuits will be performed at the frequency required by ITS.

Testing of the Unit 1 RSDP transfer and control circuits addressed in the Unit 1 Technical Specifications was completed on 12/5/95. One problem was identified during the testing that affected component operability. Specifically, the control power fuse for the "G" SRV was found blown, rendering the "G" SRV inoperable. It is not known when or why the fuse blew. It was replaced and the circuit was subsequently satisfactorily tested. This condition did not affect the "C" SRV, which is redundant to the "G" SRV. Consequently, the ability to depressurize the reactor vessel was not affected by this condition.

Regarding the 2E11-F024B valve not being fully closed, training on this aspect of the event has been included in the Beginning of Shift Training program which is attended by operations shift personnel.

ADDITIONAL INFORMATION

No similar events have occurred in the previous two years in which lack of testing of systems or a design deficiency have resulted in a loss of safety system function.

Failed Components Information:
Master Parts List Number: 2C82-S12 and S52
Type: Control Switch
Manufacturer: General Electric
Model Number: SB-1
Manufacturer Code: G080
EIIS System Code: JG
EIIS Component Code: None
Cause Code: E
Reportable to NPRDS: Yes