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Georgia Power
the southern electric system

HL-2924
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNIT 1
NRC DOCKET 50-321
OPERATING LICENSE DPR-57
LICENSEE EVENT REPORT
REACTOR COOLANT STRATIFICATION RESULTS
IN A 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 concerning thermal stratification in the reactor coolant which resulted in a condition prohibited by the Technical Specifications. This event occurred at Plant Hatch - Unit 1.

Sincerely,

J. T. Beckham, Jr.

J. T. Beckham, Jr.

JKB/cr

Enclosure: LER 50-321/1992-023

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)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	1 OF 6

TITLE (4)
REACTOR COOLANT STRATIFICATION RESULTS IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQ NUM	REV	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 8	2 7	9 2	9 2	0 2 3	0 0	0[illegible]	2 5	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL: 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (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)
[ ] 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 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	MANUFAC-TURER	REPORT TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORT TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[X] YES (If yes, complete EXPECTED SUBMISSION DATE)	[ ] NO		1 2	3 1	9 2

ABSTRACT (16)

On 8/27/92, at 0222 CDT, Unit 1 scrammed and the Group 1 Primary Containment Isolation System (PCIS) valves closed on a Main Steam Line high radiation signal as a result of organic intrusion. This event was reported in LER 50-321/92-21. During recovery from the scram transient and as a result of a lack of forced circulation, the reactor coolant became thermally stratified in the vessel. That is, relatively cold makeup water settled to the bottom head region of the reactor vessel while the upper region remained at saturation temperature. At 0430 CDT, licensed personnel noted that the vessel bottom head metal temperature was less than that allowed by Unit 1 Technical Specifications figure 3.6-2. Due to a procedure error, personnel were led to monitor the vessel metal temperature at a point above the bottom head region, which was not limiting. At approximately 1000 CDT, an orderly cooldown and vessel pressure reduction were commenced. At 0140 CDT, on 8/28/92, due to the pressure reduction, the reactor vessel metal temperature was back within the pressure/temperature limits of figure 3.6-2. At 0512 CDT, on 8/28/92, with the reactor pressure at approximately 100 psig and the reactor coolant still stratified, one Residual Heat Removal (RHR) system pump was started in the Shutdown Cooling (SDC) mode to continue the cooldown which began earlier. When the system was placed in the SDC mode, relatively t coolant was transferred into the vessel bottom head area resulting in the vessel bottom head drain temperature increasing at a rate greater than the Unit 1 Technical Specifications limit of 100 degrees Fahrenheit per hour. The cause of these events was stratification of the reactor coolant within the reactor vessel. A contributing factor was a less than adequate procedure. Corrective actions include revising procedures and training.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 1	05000321	92	023	00	2 OF 6

TEXT

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 8/27/92, at 0222 CDT, Unit 1 scrammed and the Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) valves closed on a Main Steam Line high radiation signal as a result of organic intrusion. This event was reported in LER 50-321/92-21. The Group 1 isolation resulted in closure of the Main Steam Isolation Valves (MSIVs) isolating the reactor vessel from the Main Condenser (EIIS Code SG). Immediately following the scram, the Recirculation (EIIS Code AD) system pumps tripped as designed on low reactor water level. Reactor water level was initially restored using the Reactor Feedwater (EIIS Code SK) system pumps and the Reactor Core Isolation Cooling (RCIC, EIIS Code BN) system. Several minutes into the event, Reactor Feedwater system flow was no longer available as steam is not available to the Feedwater pump turbines with the MSIVs closed. At this point, the RCIC system was primarily controlling reactor water level. The Reactor Water Cleanup (RWCU, EIIS Code XX) system had been manually isolated prior to the scram in order to perform surveillance procedure 57SV-G31-002-1S, "RWCU Differential Flow Instrument FT&C."

With both Recirculation pumps tripped, forced circulation of the reactor coolant was no longer in effect. Additionally, relatively cold water was being added to the reactor vessel via the RCIC system and the Control Rod Drive (CRD, EIIS Code AA) system. (The CRD system continuously provides makeup to the vessel bottom head region as control rod drive cooling water.) Due to these factors, reactor coolant began to undergo thermal stratification resulting in lower reactor coolant temperatures in the vessel bottom head region. Without the RWCU system in service, per the Recirculation system operating procedure, the Recirculation pumps could not be restarted. Consequently, forced circulation of the reactor coolant could not be re-established and the stratification could not be mitigated.

Following the scram, licensed personnel began monitoring the cooldown of the reactor vessel as required by procedure 34GO-OPS-013-1S, "Normal Plant Shutdown." Per the procedure, the reactor pressure and vessel metal temperature are checked every 30 minutes to ensure that the pressure/temperature limits of the reactor pressure vessel provided in Unit 1 Technical Specifications figure 3.6-2, "Pressure versus Minimum Temperature for Non-Nuclear Heatup/Cooldown and Low Power Physics Tests" are not exceeded. For monitoring vessel metal temperature, the procedure attachment used to record the data directs the individual to use point 10 on multipoint temperature recorder 1B21-R606. This point indicates vessel bottom head metal temperature.

TEXT

At 0430 CDT, licensed personnel noted that the vessel metal temperature as read from point 10 was less than that allowed by Unit 1 Technical Specifications figure 3.6-2. The thermocouple for point 10 was erroneously believed to be located in the vessel bottom head drain line. However, with RWCU secured, no flow would occur in this line which would result in the thermocouple experiencing temperatures lower than that of the vessel. Consequently, the validity of the point 10 reading was questioned. Furthermore, one step of the procedure attachment specifies the use of point 8 for plotting temperature as opposed to point 10, and would not represent the minimum vessel temperature. Point 8 indicates vessel metal temperature at a location above the bottom head. The vessel metal temperature as indicated by point 8 was within the limits of the figure. Based on this information, it was concluded that point 8 was the correct indicator to monitor. Therefore, point 8 was used for monitoring instead of point 10 and no further actions were deemed necessary.

Procedure 57SV-G31-002-1S had been stopped to allow RWCU to be returned to service. The system was warmed per procedure and, at 0850 CDT, it was returned to service. With the system back in service, flow was established through the vessel bottom head drain. However, the drain temperatures did not increase as expected. This was most likely due to the CRD system cooling flow being greater than the RWCU flow through the drain line.

Even though the RWCU system was in service, in accordance with the Unit 1 Technical Specifications, the Recirculation pumps could not be started because the dome to bottom head temperature differential was greater than 145 degrees Fahrenheit. When the drain line temperature did not increase with RWCU in service, the decision was made to begin a controlled cooldown by depressurizing the vessel in order to reduce the dome to bottom head temperature difference.

At [illegible]140 CDT, on 8/28/92, due to the pressure reduction, the reactor vessel metal temperature as indicated by point 10 was back within the pressure/temperature limits of figure 3.6-2.

At 0512 CDT, on 8/28/92, with the reactor pressure at approximately 100 psig and the reactor coolant still stratified, one Residual Heat Removal (RHR, EIIS Code BO) system pump was started in the Shutdown Cooling (SDC) mode to continue the controlled cooldown in accordance with procedure. In the SDC mode of operation, the RHR system takes a suction from the annular region of the vessel and discharges into the vessel bottom region via the Recirculation discharge piping and the jet pumps. With stratification in the vessel, the temperature of the coolant in the annular region was relatively hot in comparison to that of the coolant in the vessel bottom region. Consequently, when the SDC mode of RHR was initiated, relatively hot coolant was transferred into the vessel bottom head area resulting in the vessel bottom head drain temperature increasing from 90 degrees Fahrenheit to 310 degrees Fahrenheit in 10 minutes. The temperature change over an hour was from 90 degrees to 310 degrees Fahrenheit. This was in excess of the [illegible] degrees Fahrenheit per hour heatup limit specified in Unit 1 Technical Specifications section 3.6.A.

TEXT

CAUSE OF EVENT

The cause of the decrease in vessel bottom head metal temperature below the limit of Unit 1 Technical Specifications figure 3.6-2 and the subsequent increase in temperatures in this region of greater than 100 degrees Fahrenheit per hour was thermal stratification of the reactor coolant within the reactor vessel. Specifically, coolant at the top region of the vessel was at saturated temperature and within the limits of figure 3.6-2, while the coolant in the bottom region of the vessel was significantly lower and outside the limits of figure 3.6-2 for the reactor pressure which existed. Furthermore, when the SDC mode of RHR was initiated, the relatively hot coolant in the annular region of the reactor vessel was transferred to the relatively colder bottom head region resulting in the temperature of the bottom head region increasing at a rate greater than that allowed by Unit 1 Technical Specifications section 3.6.A.

The fact that stratification occurred was largely the result of the scram transient itself and the RWCU system being out of service at the time of the scram. Immediately following the scram, the Recirculation pumps tripped, as designed, resulting in no forced circulation of the reactor coolant. The relatively cold makeup water from the RCIC and CRD systems, because of its higher density, migrated to and/or remained in the bottom head region of the vessel. This is a recoverable situation if RWCU can be placed in service expeditiously and the Recirculation pumps restarted. However, in this event, RWCU had been removed from service for surveillance purposes two and a half hours prior to the event. Consequently, the system had to be prewarmed before being placed in service and, therefore, could not be returned to service immediately. Without RWCU in service, as required by the Recirculation system operating procedure, the Recirculation pumps could not be started. By the time RWCU was warmed up and placed in service, the 145 degree limit for starting the Recirculation pumps was exceeded, further precluding restart of the pumps. The capacity of the RWCU system was not sufficient to remove the cold water from the vessel bottom head region. Thus, a controlled vessel cooldown and consequent reduction in vessel pressure had to be initiated in order to decrease the vessel top to bottom head temperature differential to within the 145 degree limit and allow restart of the Recirculation pumps. The initiation of the shutdown cooling mode of RHR was part of the controlled cooldown effort.

It is recognized that an error did exist in procedure 34GO-OPS-013-1S as previously discussed. This error led operators to monitor vessel metal temperature at a point above the bottom head region, which was not limiting. While this error did not cause operation outside the limits of figure 3.6-2, it did apparently contribute to delays in reducing reactor pressure and restoring operation within the limits of the curve.

TEXT

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73(a)(2)(i) because a condition existed that was prohibited by the Technical Specifications. Specifically, following a plant transient, the temperature in the bottom head region of the reactor pressure vessel decreased below the limit allowed by figure 3.6-2 of the Unit 1 Technical Specifications. Additionally, when the SDC mode of RHR was initiated, the temperatures in the bottom head region of the vessel experienced a heatup in excess of the 100 degrees Fahrenheit per hour limit addressed in Unit 1 Technical Specifications section 3.6.A.

3.6-2 of the Unit 1 Technical Specifications contains the reactor vessel pressure/temperature limits which are based on the fracture toughness analysis of the vessel for non-nuclear heatup/cooldown as well as for low power physics testing. This figure specifically reflects those limits which apply to the core beltline material. They are more conservative than those for other areas of the vessel because of the postulated embrittlement of the beltline region resulting from neutron exposure to the region.

In this event, the vessel bottom head metal temperature exceeded the pressure/temperature limits of figure 3.6-2. As noted earlier, these limits were developed for the beltline region and are more limiting than those for the vessel bottom head which is remote from the beltline. General Electric reviewed the event against the limits for the vessel bottom head region and found that the temperature limits were exceeded by 15 degrees Fahrenheit at one point during the event. An assessment of the safety factors associated with these established limits showed that even in exceeding the limit by 15 degrees, substantial safety margin still existed. However, in order to demonstrate strict compliance with the safety margin of ASME Code Section III, Appendix G, a more detailed analysis is being performed by General Electric.

The Technical Specifications heatup/cooldown rate limit of 100 degree Fahrenheit in one hour was assumed as being the normal heatup/cooldown rate in analyzing the temperature and pressure limits for the reactor pressure vessel. The stress intensity and fatigue limits experienced at this rate were analyzed and found to be within the requirements of Section III of the ASME Boiler and Pressure Vessel Code. In this event, the vessel bottom head region experienced a heatup in excess of 100 degrees in one hour. An analysis of the heatup transient by General Electric showed that the resultant stress on the vessel bottom head meets the appropriate ASME Code limits. Additionally, it was determined that the fatigue impact of the heatup event on the vessel bottom head was acceptable due to the substantial fatigue margin associated with the head region.

Based on this information, it has been determined that this event did not present a serious challenge to the integrity of the reactor pressure vessel. Therefore, this event had no effect on public health and safety.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5) YEAR	SEQ NUM	REV	PAGE (3)
PLANT HATCH, UNIT 1	0 5 0 0 0 3 2 1	9 2	0 2 3	0 0	6 OF 6

TEXT

CORRECTIVE ACTIONS

The causes and effects of vessel coolant stratification following a loss of forced circulation will be discussed with operations shift personnel in beginning of shift training. This will be completed by 10/.1/92.

Procedure 34GO-OPS-013-1S has been temporarily revised to correctly show which points of recorder 1B21-R606 are to be monitored during heatup/cooldown. The temporary revision will remain in effect until the procedure is permanently revised. This action will be completed by 11/14/92. The Unit 2 procedure was reviewed and found to be deficient in this regard also and will be revised prior to startup from the current refueling outage.

As noted previously, an event specific analysis per the ASME Code will be performed by General Electric. The analysis is scheduled to be completed by 11/30/92. The results of this analysis will be included in an update to this LER which will be submitted by 12/31/92.

ADDITIONAL INFORMATION

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

No previous similar events have been reported in the past two years in which the reactor coolant heatup/cooldown rate or the reactor pressures and temperatures resulted in a condition prohibited by the Technical Specifications.

No failed components contributed to or resulted from this event.