Georgia Power Company
333 Piedmont Avenue
Atlanta, Georgia 30308
Telephone 404 526-6526

Mailing Address:
Post Office Box 4545
Atlanta, Georgia 30302

L. T. Gucwa
Manager Nuclear Safety
and Licensing

Georgia Power
the southern electric system

SL-4187
0137I
X7GJ17-H800

March 14, 1988

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

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
SPECIAL REPORT 88-004
REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

Gentlemen:

Georgia Power Company is submitting the enclosed, voluntary, Special Report (SR) concerning an event where the cooldown rate of the reactor was briefly exceeded. This event occurred at Plant Hatch - Unit 2.

Sincerely,

L. T. Gucwa

L. T. Gucwa

LGB/lc

Enclosure: SR 50-366/1988-004

c: (see next page)

IE22
1/1

Georgia Power

c: Georgia Power Company
Mr. J. T. Beckham, Jr., Vice President - Plant Hatch
GO-NORMS

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. P. Crocker, Licensing Project Manager - Hatch

U. S. Nuclear Regulatory Commission, Region II
Dr. J. N. Grace, Regional Administrator
Mr. P. Holmes-Ray, Senior Resident Inspector - Hatch

0137I

Georgia Power

ENCLOSURE

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
SPECIAL REPORT 88-004
REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

REQUIREMENT FOR REPORT

This Special Report is being submitted because the event may be of interest to other members of the nuclear industry relative to lessons learned. The event occurred when Plant Hatch Unit 2 was shutdown, entering its seventh refueling outage.

The event was evaluated as not being reportable per the requirements of 10 CFR 50.73 (a)(2)(i) (condition prohibited by the plant's Technical Specifications) because all the actions required by the applicable Technical Specifications action statement were met within the required time periods. As such, the requirements of the Technical Specifications were met and not violated.

SEQUENCE OF EVENTS

On 1/13/88, the unit was being brought to a hot shutdown condition by inserting control rods into the reactor core. (At an appropriate power level, a scram signal would be inserted into the Reactor Protection System (RPS) circuitry.) The feedwater system was configured such that a reactor feedpump, and a condensate booster pump and a condensate pump were in service.

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

As part of the normal shutdown operations that were underway, the main turbine and generator had been removed from service, resulting in the heating source for all feedwater heaters being removed (these heaters heat the feedwater using steam from the main turbine). Feedwater temperature was approximately 100 degrees Fahrenheit (^{o}F).

At 0754 CST, the reactor was at an approximate power level of 13 percent of rated thermal power. At this time, the plant operations personnel inserted a manual scram signal into the RPS logic. (This would bring the reactor subcritical and start the refueling outage.)

As expected, subsequent to the scram, reactor water level decreased to approximately +22 inches (approximately 186.4 inches above the Top of Active Fuel - TAF). The reactor feedpump responded to the decrease in the reactor water level and started to inject water into the reactor vessel.

Since this feedwater was cooler than normal (due to the fact that the feedwater heaters were not receiving steam flow), there was a decrease in the reactor pressure to approximately 700 psig.

The reactor feedpump had recovered reactor level and had tripped off on a high reactor water level signal, per design. In order to maintain reactor water level, one of the reactor feedpumps was restarted when reactor water level had decreased sufficiently.

Plant operations personnel placed the mechanical vacuum pump into service in order to remove the Steam Jet Air Ejectors (SJAE). This allowed the Main Steam Isolation Valve (MSIVs) to still be open after the scram and the main condenser could be used as a heat sink. This method of operation is utilized to better control the cooldown rate of the reactor.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

At approximately 0830 CST, the reactor feedpump (which had been placed into service to control reactor water level), tripped, per design, on a high water level signal. At this time, operations personnel observed the water level indicators. The narrow range indicators (0 to +60 inches) were indicating reactor water level in the range of +58 to +60 inches. The wide range shutdown flooding level indicator (which indicates above +60 inches) was inoperable. It was not required to be operable. While this instrument is not accurate when the reactor is in power operations, it can be used to detect gross changes in reactor water level.

Plant operations personnel recognized and anticipated that when the reactor feedpump injected into the reactor, the water level indication would be in the upper ranges of the level instruments. This is a normal occurrence. Plant personnel were anticipating that the level indications would return to the lower instrument ranges after the water level had stabilized.

The operators also observed that the level recorder was indicating a level of approximately +58 inches (while the level instruments were indicating level of approximately +60 inches). From this information, the plant operators believed that reactor water level was in the range of +58 to +60 inches. However, operations personnel did not know at this time that the reactor water level recorder was stuck, erroneously indicating an on-scale level of +58 inches.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

Operations personnel also did not know at this time that the startup level control valve, 2N21-F111, was leaking. The indication in the main control room showed that this valve was fully closed. Due to the recent injection of cool water (by the reactor feedpump), the reactor pressure had decreased below the shutoff head of the condensate booster pump. This then allowed the condensate booster pump to inject water into the reactor vessel past the leaking startup level control valve.

In the period of time from 0833 to 0900 CST, the shift supervisor was notified that the cool down rate was approximately 75°F/hr and increasing as measured on the recirculation loop suction temperature instruments. The operating staff was directed to increase monitoring of the cooldown rate. The cooldown rate continued to increase. When the cooldown rate reached approximately 95°F/hr, the shift supervisor ordered the main steam isolation valves closed. It was in this period of time (from the start of the scram at 0754 CST until approximately 0900 CST) that the cooldown rate of 100°F in any one hour was exceeded. Based on the saturation temperature at the steam dome pressure, the maximum cooldown rate was approximately 104°F/hr.

With the MSIVs closed, reactor pressure increased until the reactor pressure equaled the shutoff head of the condensate booster pump. At this pressure, no more water was being injected into the reactor vessel through the leaking startup level control valve, and the cooldown rate was reduced to less than 100°F in any one hour. As a result of exceeding the cooldown rate, operations personnel recognized that they had entered a Limiting Condition for Operations (LCO).

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

Operations personnel continued with the shutdown. At 0906 CST, they opened a Reactor Water Clean Up (RWCU) valve to allow the reactor water level to decrease by flowing into the main condenser. At 0924 CST they documented the fact that they had entered an LCO condition. They later secured the condensate booster pump.

EVENT REVIEW

The event was reviewed to determine the causes of the event and to determine if there was any adverse impact on plant equipment. It was determined that the cause of the cooldown was primarily the result of equipment failure, (specifically, the failure of startup level control valve 2N21-F111 to completely close) combined with the thermodynamic state of the reactor coolant.

The startup level control valve allowed water to flow either because of a faulty (blocked) orifice in a pneumatic control relay or because of valve seat leakage. With this orifice blocked, the valve could not completely close, even though the controller was demanding 100 percent closure.

The thermodynamic state of the reactor coolant was a function of the power level at the time of the scram and the temperature of the feedwater. Since the scram occurred at a relatively low power level, there was little decay heat and the feedwater was cooler than normal since there was no feedwater heating (the main turbine had been taken off line a few hours prior to the reactor manual scram). These two factors combined to result in a higher than normal depressurization rate for the reactor vessel. This allowed the booster pump to force water past valve 2N21-F111 into the reactor. This occurred until the reactor pressure equaled the shutoff head of the condensate booster pump.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
<u>BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS</u>

The Unit 2 Technical Specifications, section 3.4.6.1 states:

> "The reactor coolant system temperature and reactor vessel pressure shall be limited . . . for . . . cooldown following a nuclear shutdown . . . with . . . a maximum cooldown of 100°F in any one hour period.
>
> With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least hot shutdown within 12 hours and in cold shutdown within the next 24 hours."

The unit was in a hot shutdown condition at the onset of the event. However, when plant operations personnel determined that the cooldown rate had been exceeded, they returned the out of requirements parameter to the acceptable range within the 30 minutes specified by the plant's Technical Specifications.

The action statement also required that an engineering evaluation of the event be performed. Representatives of the Nuclear Steam Supply System (NSSS) vendor (General Electric - GE) were contacted for the evaluation and were requested to provide the basis for the Technical Specifications as reflected in the Reactor Pressure Vessel (RPV) design. This assured that the Technical Specifications had not been misinterpreted.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

The basis for the Technical Specification, according to GE, is a vessel cooldown of 100°F in any one hour period, commencing at the initiation of the cooldown and measured at 30 minute intervals. The temperature utilized is the saturation temperature associated with the vessel pressure during the cooldown. The use of the recirculation loop suction temperatures (the temperatures that the operators used to monitor the cooldown rate) will provide an overly conservative estimation of the cooldown rate.

GE performed an engineering evaluation of the event and presented their findings in a report dated 1/13/88. The evaluation reviewed the event for structural concerns in three areas: 1) brittle fracture, 2) allowable stresses, and 3) fatigue.

In order to bound the event, the engineering analysis compared the event with a Single Relief Valve Blowdown (SRVB) event. The SRVB event was based on a cooldown rate of approximately 1026°F/hr. By using this cooldown rate, it was determined that the event that occurred on 1/13/88 was completely bounded by the SRVB event.

With regard to the three areas of concern (brittle fracture, allowable stresses and fatigue), the report evaluated each condition. For the consideration of brittle fracture, the report examined the ranges of pressures and temperatures that were recorded in the event and compared these ranges with a postulated 145°F/hr cool down rate (maximum change in recirculation suction temperature). Additionally, the American Society of Mechanical Engineers (ASME) Appendix G analysis was also reviewed. This analysis was used to develop the curves of cooldown rates that appear in the Technical Specifications. These limits, have a large degree of margin against fracture. Based on this information and the analysis of the ranges of pressure and temperature that occurred in the event, it was concluded that the transient created no concern of brittle fracture of any vessel components.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

The report noted that the event did create some cooldown stresses in the vessel components. While the cooldown did exceed the limit of 100°F in any one hour, this limit was not used in the analysis of the vessel. The vessel was analyzed for a Single Relief Valve Blowdown (SRVB) and the SRVB event is much more limiting than the event that was actually experienced on 1/13/88. Based on this information, from an allowable stress standpoint, the event did not have any adverse impact on the reactor vessel.

The vessel was also analyzed for the impact of fatigue in light of the SRVB event. As a conservative assumption (to bound the analysis) the event that occurred on 1/13/88 was classified as a SRVB event for fatigue analysis purposes. As such, the report concluded that the event of 1/13/88 did not have any adverse impact from a fatigue perspective.

The engineering report determined that event's impact is acceptable and created no concerns. The analysis concluded that " . . . there are no structural integrity concerns with continued operation of the Hatch Unit 2 vessel."

To further confirm the GE findings, plant personnel calculated the changes in the Cumulative Fatigue Usage Factors (CFUFs) for the Reactor Pressure Vessel Main Closure Studs and the Reactor Pressure Vessel Shell. The CFUFs for the main closure studs prior to this event was 0.215. After the event, the CFUF was calculated as 0.225. The CFUF for the vessel shell prior to the event was 0.019. After the event, the CFUF was calculated as 0.019167. The limit for both of the CFUFs is 1.0.

Georgia Power

ENCLOSURE (Continued)

REACTOR COOLDOWN RATE EXCEEDS 100 DEGREES FAHRENHEIT IN AN HOUR
BUT IS RECOVERED WITHIN TECHNICAL SPECIFICATIONS LIMITS

Additionally, during this event, it was determined that since the condensate booster pump continued to inject into the reactor vessel, there was the possibility that some water entered the Main Steam Lines (MSL) prior to the closure of the MSIVs. (However, it is concluded that no water entered the MSL downstream of the outboard MSIV prior to the closure of the MSIVs.) The impact of the potential filling of the steam lines was examined in greater detail by representatives of GE.

GE evaluated three cases, based on differing plant conditions. These Thermal Cases were: 1) Reactor Pressure Vessel (RPV) at 540°F, main steam piping at 540°F and the piping full of steam (normal operating condition); 2) RPV at 540°F, main steam piping at 430°F and full of water; and 3) RPV at 540°F, top riser and horizontal run at 430°F and the lower leg of piping to the penetration at 150°F, with the piping full of water.

The analysis was performed using one of the main steam lines (the D line). This line was chosen because it was found that this line envelopes the other three steam lines. When this line was analyzed for the above three cases, it was determined that the fatigue analysis results were in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1979 edition. Based on this information, it was concluded that " . . . the stresses in the piping are within the code allowable and the loads on the reactor pressure vessel nozzle and hangers are within their acceptable limits."