for replacement of several CR 120 relay~~s~~ coils. He received permission from the "extra" Unit 2 SS to enter the panel.

11:31 a.m. The engineer moved a wire bundle to verify a wire number. He observed an "arc" or "flash" and notified the Unit 2 SS within a minute. The SS and the engineer inspected the back panel and then the SS directed the operators to walkdown their panels. Almost immediately it was noted that the mechanical vacuum pump indicated tripped.

Subsequent investigation indicated that fuse 2A71-F22 blew. A partial Group I and Group II isolation occurred. The 2E11-F015B valve, B SDC injection valve to the reactor recirculation loop went shut.

Reactor water level (actual) was 37 inches and RWCU HX inlet temperature was 168 degrees F prior to the valve going closed. Both loops of core spray and all three EDGs were operable. Also, the SRVs were available.

11:40 a.m. Operators identified that F015B valve was shut and attempted to reopen it. ~~Several times~~ The valve was reopened twice but immediately shut. Operators also tried, but unsuccessfully, to reset the isolation signal. The 2B RHR pump was secured and the 2E11-F017B valve was shut.

Operators entered 34AB-E11-001-2S: Loss of Shutdown Cooling. Concurrently, electricians are contacted to investigate the cause of the isolation.

11:~~45~~ 58 a.m. Operators began to raise reactor water level, using the CRD system in accordance with step 4.6 of 34AB-OPS-001-2S in order to ensure a natural circulation flowpath in the reactor vessel. Level was increased to 57 inches (actual).

Electricians did not identify the blown fuse in panel 2H11-P623 on their initial check. Fuse F22 is located in an enclosure within the panel. The SS verified that the breaker to the ~~F015B valve~~ RHR "B" logic was closed.

12:00 p.m. Reactor pressure increased to 1.4 psig. The pressure increase was not known by the operators since, in accordance with procedures, they were monitoring indicators 2C32-R605A, B, and C. These indicators, on the CR panels, have a scale of 0-1200 psig. Additionally, the reactor head vents were already open. Review of computer information, after the event, supplied the information of pressurization.

12:02 p.m. SS noted that RWCU inlet temperature was increasing and restoration of the "B" SDC loop would apparently be delayed. Additionally, the SOS noted that some reactor vessel metal temperatures were increasing. He directed that the "A" loop of RHR be placed in SDC. Procedure 34SO-E11-010-2S: RHR system is followed. The system flushing was omitted under the direction of the SS. The procedure was temporarily changed in accordance with TS requirements. The F015B valve was not opened manually due to concerns about overriding an ESF.

Increased monitoring of temperatures and pressure in accordance with Attachment 1 of procedure 34GO-OPS-015-2S was performed as directed by 34AB-E11-001-2S.

12:30 p.m. One of the resident inspectors entered the control room on a routine tour and observed operator actions.

12:54 p.m. SDC flow through the vessel with the "A" RHR HX was established. A maximum temperature of 195 degrees F was observed on the RWCU HX inlet as flow was restored.

1:25 p.m. Fuse 2A71-F~~0~~22 was restored and the isolation was reset.

1:55 p.m. The NRC headquarters duty officer was informed of the event in accordance with 10 CFR 50.72 (b) (2) (ii). Region II management had been previously contacted by the resident inspectors.

2:20 ~~2:00~~ p.m. In accordance with procedures for reactor disassembly, the flange for a reactor vessel head instrumentation line was unbolted by pipefitters in the reactor cavity. Steam was observed as the bolts are loosened. After the bolts are removed, an approximate 2-3 feet long plume of steam came out of the pipe. The steam is observed coming out of the line for approximately 20 minutes.

Management directed that no work or entry into critical CR panels would be permitted until the reactor cavity is flooded. Additionally, scheduled work on the "A" RHR loop was delayed until after the cavity is flooded.

4. Equipment Performance (71707)

The specific cause of the fuse blowing when the wiring was moved could not be determined without a close examination of the panel in which the fuse and relays are located. Because the irradiated fuel was still being offloaded from the vessel at the end of this report, examination of the panel was not completed. It would not be prudent to enter the panel until the fuel is completely offloaded. When conditions permit, the inspectors and the licensee's event review team will examine the interior of the panel and the associated wiring.

 process

The inspectors' reviews indicated that after the initiation of the event, all safety significant plant equipment performed as required. Indications, including SPDS and the ~~plant~~ computer, were available and functioning. The restoration of shutdown cooling by startup of the "A" RHR loop was not delayed by any equipment problems.

5. Personnel Performance (71707) (93702)

One of the inspectors observed some of the recovery actions in the CR. The inspector noted that actions were timely and were accomplished in accordance with the procedures. The decision to omit flushing of the "A" RHR loop prior to restoration of flow to the vessel was appropriate. A temporary procedure change to allow omission of the flushing was completed as permitted by TS. The inspectors noted that conductivity increased from 2 micromhos/cm to 8 micromhos/cm as a result of the "A" RHR loop flow. The limit on vessel conductivity during shutdown is 10 micromhos/cm. The inspectors reviews of logs and other records also indicated that procedures had been followed and appropriate actions were taken. Specifically, all applicable steps of the Loss of Shutdown Cooling procedure were verified to have been completed.

6. Reactor Pressurization and Vessel Head Vent Issues (71707) (37700) (40500)

a. Reactor Pressurization During Temperature Increase

Early in the investigation of the event, it was noted that one process computer point (B025) indicated that the reactor had become pressurized during the event. This computer point recorded pressure as sensed from transmitter PT 2C32-N005B. This transmitter is connected to piping from water level reference leg 2B21-D004A. The data indicated a slow and steady increase in pressure from an offscale low value at about 11:30 a.m. to a peak of ~~8.3 psig at about 12:48 p.m.~~ After the SDC flow was restored, the pressure decreased. 9.1 psig at 12:51 p.m.

The pressurization was not recognized by the CR operators since they were monitoring other 0-1200 psig indicators as permitted by procedures. Additionally, the vessel head vents were open which indicated to many personnel that the vessel would not pressurize. The procedure permits monitoring of the computer indications but that option was not chosen.

Investigation was conducted into the indication of reactor pressurization. The indication was supported by analysis and other information:

- GE calculated that 5-8 psig of pressure could have been present in the reactor after 1.5 hours without shutdown cooling flow. The calculation assumed 30 hours after shutdown, 70 percent RTP, CRD in operation, and reactor venting through 100 feet of 1.0 inch piping. These conditions were very close to those present at Hatch during the loss of SDC. The inspectors requested this calculation for NRC review.

- Testing of the reactor pressure instrumentation, which is indicated by computer point B025, demonstrated that the instrument is accurate and within calibration.

- It was also confirmed that at about ~~2:00~~ 2:20 p.m. (approximately 1 hour and 20 minutes after SDC flow was restored) steam was emitted during the breaking of a flanged instrumentation connections on the vessel head. A 2-3 feet long steam plume came out of the connection as it was unbolted.

The inspectors observed portions of a test which was conducted to verify that the head vent path was open. Water flowed through the path from upstream of the vent valves to the drywell sump. Additionally, the inspectors and the ERT verified from drawings that the backpressure resulting from a full sump on the line would be limited to less than two feet of water elevation. The sump is vented to the drywell. Also, no increase in sump temperature was noted during the event.

The inspectors concluded that the temperature increase following the loss of shutdown cooling and boiling of the water in the region of the core resulted in a pressurization in the vessel. (Paragraph 7 of this report addresses the temperature increase.) The reactor pressure increase was unexpected and not recognized by the operators primarily due to procedural weaknesses.

b. Vessel Head Vent Issues

A contributing factor in this event was that numerous operators and other personnel felt that with the reactor vessel head vents open, the reactor would not pressurize in such an event. Procedure 34AB-E11-001-2S: Loss of Shutdown Cooling, contains a caution which stated that vessel pressurization could occur if vessel flow is less than 7700 gpm and level is less than or equal to 53 inches, if the vessel is not vented. Procedure 52GM-MME-015-2S: Reactor Vessel Disassembly, required that before the head piping flanges are disconnected, the CR is contacted to confirm the vessel is vented by verifying that the vent valves are open. The ERT and the inspectors questioned a number of operators who stated that they had thought pressurization would not occur with the vents open.

The inspectors reviewed available documentation in an effort to confirm the function of the vent valves and the vent path. The inspectors noted that Section 5.1 of the Unit 2 FSAR stated that the radwaste system (drywell sump) provides a collection point for gas and vapor venting from the RPV during RPV heatup. It also stated that the small size of the vent valves (the valves are located in piping which is ½ inch in diameter) and the fact that the piping is attached to a vented sump make it unlikely that a failure of the administrative controls on the vent valves would result in radwaste system overpressurization.

The inspectors reviewed a February 26, 1980 letter from the NRC to GPC which addressed the function of the vent valves. The letter documented the NRC's evaluation of the licensee's compliance with category "A" items concluded that it was fortuitous that an inadvertent change in Operational Conditions did not occur. The deficiencies are directly related to plant conditions of a high decay heat load.

The available temperature indications at Hatch after cooldown are primarily sensed from process lines connected to the vessel via the reactor recirculation lines. Recirculation pump suction, RWCU heat exchanger inlet, or RHR heat exchanger inlet temperatures are monitored depending on plant conditions. Procedure 34AB-E11-001-2S: Loss of Shutdown Cooling contains several statements which indicate that if reactor water level is maintained high enough (without forced flow), natural circulation in the core will result in adequate mixing such that monitored temperature indications will accurately reflect temperatures in the core region:

- A caution stated that if reactor water level is less than or equal to 53 inches and vessel flow is less than 7700 gpm, coolant heatup may occur in the core area with no indicated temperature increase at the recirculation pump suction or RWCU inlet and that pressurization could follow if the vessel is not vented.

- A note stated that if SDC is not in service and level is not greater than 53 inches, a recirculation pump can be periodically started to provide mixing and accurate temperature measurement.

- Step 4.7 required increased monitoring of temperatures and pressures to at least 15 minute intervals using the recirculation loop, RWCU, or RHR heat exchanger temperatures.

The inspectors concluded that the procedure does not inform operators that even with high reactor levels, the indicated temperatures may not be indicative of core area temperatures. TS define Operational Condition 3 as average reactor coolant temperature greater than 212 degrees F. Since additional actions are contingent on reactor water temperatures and operators are expected to make decisions based on reactor temperatures, the procedure should provide more guidance. In this instance, the operators observed RWCU heat exchanger inlet temperatures of 195 degrees F and average reactor coolant temperature was subsequently calculated to be ~~approximately~~ between 195 and 210 degrees F.

On 3/21/94, GE calculated a value for average or bulk RCS temperature during the event to be approximately 210 degrees F. TS limits refer to RCS average temperature of 212 degrees F. When shutdown cooling flow was restored, the temperature at the inlet to the RHR HX peaked at 195 degrees F. While it is known that some quantity of water in the core area of the vessel was above 195 degrees F, the average temperature of the RCS is not clear.

In accordance with step 4.7.2 of procedure 34AB-E11-001-2S, reactor vessel metal temperatures were monitored. This information contributed to the decision to restore SDC flow by starting of the "A" RHR loop. The restored, steam was emitted for several minutes. Additional verification that the reactor is not pressurized prior to piping disassembly may be appropriate.

Two apparent violations were identified.

8. Previous Loss of Shutdown Cooling Events Incidents (40500) 92700) (92702)

There have been several other loss of shutdown cooling events at Hatch, some of which were similar to this event. The previous events did not occur with high decay heat loads present. The inspectors noted that three cases involving fuse A71-F___ and the E11-F015 valve have occurred since 1986.

LER 321/86-17: Personnel Error Cause Loss of Shutdown Cooling, addressed an instance in which inadvertent grounding during logic system functional testing resulted in the 1E11-F015B valve going shut and flow being lost for three hours prior to detection. Fuel was in the vessel and the reactor head was in place. The "A" loop of RHR was inoperable. Because decay heat was low at the time of the event, it was considered of low safety significance. The safety assessment stated that the F015B valve could have been manually opened if necessary.

On April 14, 1993, shutdown cooling flow was interrupted on Unit 1 for about 3 hours. In this case, the cavity was flooded and connected to the SFP. A small amount of fuel had been reloaded into the core. During, removal of a panel board resulted modification work on a control room panel (front panel) ~~manipulation of some~~ indication wiring in the panel contacting ground, causing ~~caused~~ fuse 1A71-F022B to blow and valve 1E11-F015B to shut. IR 50-321,366/93-06 describes the inspector's review of that event. Violation 50-321/93-06-01: Failure to Comply with Shutdown Cooling TS Requirements, was issued. A management meeting was held in Region II to discuss the event, particularly the long period of time (1.5 hours) before the loss of SDC flow was identified.

The corrective actions for the April 1993 event emphasized methods to improve the recognition of a loss of SDC flow and ensure that personnel pursue any indications of grounding of electrical circuits. The inspectors noted that several of these corrective actions were effective in that the engineer immediately informed the operators of the problem. An additional action was to establish a computer alarm which would be actuated if core flow decreased below a certain value. The flow signal was developed from jet pump differential pressure signals. This alarm was established and functionally tested just after RHR was placed in service earlier in the outage. During the test, when the RHR pump was shutdown, the jet pump differential pressure decreased below the alarm setpoint and the alarm was actuated. However, during the loss of SDC incident, this alarm did not actuate. Review of the chart recorders indicated that the jet pump differential pressure did not fall below the alarm setpoint. The licensee is still evaluating the implementation of design modifications to provide an alarm on loss of RHR flow. The ERT, after the April 1993 incident, recommended the modification.

As discussed in paragraph 7 of this report, the inspectors concluded that the procedure for monitoring of plant conditions while in cold shutdown did not appropriately address conditions of high decay heat loading. While the frequency of monitoring checks was increased and was intended to more rapidly alert operators to a loss of shutdown cooling flow, this event demonstrated that the frequency was not sufficient under high decay heat conditions.

The initiating cause of the April 1993 event was very similar to the latest incident. In both cases, work in a control room panel resulted in inadvertent grounding of some PCIS logic and fuse A71-F022 blew. After reviewing this incident, the inspectors concluded that the actions of the engineer or the SS who permitted access to the panel were not unreasonable. In general, the licensee had been controlling activities involving the circuitry for the division of RHR that was in service. As discussed above, the inspectors concluded that the licensee could have been more conservative in regards to RHR system outages prior to cavity floodup.

The other previous loss of shutdown cooling events consisted primarily of instances in which the shutdown cooling suction valves were shut. The inspectors reviewed INs and NRC Bulletins related to decay heat removal issues. The documents related to BWR events were directed at inventory losses due to DHR system breakers or inadvertent valve operations. Additionally, NUREG-1449: Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States was referenced during the inspectors review of this event.

9. Outage Planning Issues (71707) (40500)

The inspectors' reviews concluded that the loss of shutdown cooling flow itself was not reportable and would not cause the licensee to enter emergency event classifications. The major concern is the reactor pressurization without primary containment. Additionally, this event may point out the necessity for increased monitoring of reactor vessel and core conditions and/or different venting systems.

The incident demonstrated the vulnerability of BWR-4 shutdown cooling systems. Although some TS do not permit an inoperable train of SDC until the reactor cavity is flooded and connected to the SFP, Hatch TS do not contain that restriction. This was prominently noted in IR 50-321,366/93-06, which addressed a previous loss of SDC at Hatch.

An additional concern, as noted in IR 50-321,366/93-06 and discussed during the management meeting, was that the licensee's outage schedule places the plant in a vulnerable position regarding shutdown cooling systems. Prior to the reactor cavity being flooded, with relatively high decay heat loading, one loop of RHR is removed from service for maintenance activities. While this is permissible by Hatch TS, under high decay heat loads, these actions are not conservative. Information in SIL 357 also recommends that RHR trains not be removed from service prior to flooding the cavity. both

The inspectors confirmed that if the incident had not occurred, the "~~B~~ A" SDC loop would have been rendered inoperable for LLRT testing sometime later on March 17. However, both trains of CS would have been available. Also, the reactor head was detensioned at about 1:00 a.m. on March 18.

During discussions with personnel involved in the vessel disassembly work, it was noted that the licensee's practice is to maintain water level below the level of the main steam lines for as long as two days after shutdown. The level is maintained to complete LLRT of the MSIVs. The inspectors noted that additional inventory in the reactor vessel would assist in RPV metal cooling and provide an increased time to boiling in a loss of shutdown cooling event.

The reactor cavity was flooded up and connected with the SFP at about 2:00 pm on March 19. Currently the "B" loop of SDC and the "B" loop of CS are operable. A supplemental decay heat removal system is being tested.

10. Significance and Regulatory Issues of Event

Chapter 15 of the Unit 2 FSAR contained a summary of an analysis for a loss of RHR Shutdown Cooling. The FSAR stated that no cladding temperature increase will occur, because boiling transition will not be reached. Assuming the highest decay heat loading at only four hours after shutdown, nucleate boiling heat transfer will be maintained and MCPR will remain high.

As discussed in paragraph 9 above, additional systems were available to maintain water inventory in the core. If the "A" loop of RHR had not been available, both trains of core spray were available and with the SRVs could have been used as an alternate decay heat removal system. If the incident occurred with the RPV head piping removal, (pressurization for SRV actuation not available) CS would still be available as an injection source.

Both sources of offsite power were operable during the event. The "A" EDG was removed from service several hours after the event, but 2 EDG's remained operable to supply Unit 2 loads.

The inspectors concluded that the safety significance of the incident regarding potential release of fission products or fuel damage was mitigated by the above available system. The significant concerns involve the possibility of inadvertently changing the operational condition of the plant and the unrecognized pressurization.

Although the drywell was breached, secondary containment was maintained on the refueling floor. Unit 2 DW had been established as part of Unit 1 secondary containment. The inspectors noted that the Unit 2 RB was open to atmosphere (the RB doors were open) and contained systems connected to the RCS. The inspectors also concluded that if temperature would have reached 212 degrees F, containment could not have been restored within the TS action statement time period. The licensee's action most likely would have been focused on reducing temperature below 212 degree F. Unit 2 TS 3.0.4 stated that entry into an operational condition shall not be made unless the LCO conditions are met without reliance on action statement provisions. If temperature would have exceeded 212 degree F, a TS violation would have occurred.

The NRC has not completed a review of the licensee's or GE's calculation which concluded that the average reactor coolant temperature was 210 degrees F.
between 195 and

As discussed in NUREG-1449, since the vessel head was still tensioned during the event, a large steam release into secondary containment would not be expected to occur. If the licensee would have proceeded with the outage plans, the vessel head would have been removed a few hours after the time of the event and the other train of RHR would have been inoperable. Under those conditions, it is postulated that boiling would occur within 2 hours and the steam could affect secondary containment.

11. Exit Interview

The inspection scope and findings were summarized on March 25, 1994, with those persons indicated in paragraph 1 above. The licensee did not identify as proprietary any of the material provided to or reviewed by the inspectors during this inspection.

Item Number	Status	Description and Reference
50-366/94-09-01	Open	(Apparent Violation)- Shutdown Cooling Procedures Inadequate for High Decay Heat Conditions, paragraph 7.
50-366/94-09-02	Open	(Apparent Violation) Cold Shutdown Monitoring Procedure Not Corrected to Frequency Commensurate With Decay Heat Load, paragraph 7.

12. Acronyms and Abbreviations

AGM-PO- Assistant General Manager - Plant Operations
AGM-PS- Assistant General Manager - Plant Support
BWR - Boiling Water Reactor
CFR - Code of Federal Regulations
CR - Control Room
CRD - Control Rod Drive
CS - Core Spray
DC - Deficiency Card
DHR - Decay Heat Removal
DW - Drywell
ECCS - Emergency Core Cooling System
EDG - Emergency Diesel Generator

ENCLOSURE 3

LIST OF ATTENDEES

U. S. NUCLEAR REGULATORY COMMISSION

E. Merschoff, Acting Deputy Regional Administrator, Region II (RII)
J. Johnson, Acting Director, Division of Reactor Projects (DRP), RII
D. Matthews, Director, Project Directorate II-3 (PD II-3), Office of Nuclear Reactor Regulation (NRR)
M. Sinkule, Chief, Reactor Projects Branch 3, DRP, RII
P. Skinner, Chief, Reactor Projects Section 3B, DRP, RII
B. Holbrook, Resident Inspector, Hatch, DRP, RII
D. Wheeler, Project Manager, NRR
T. Peebles, Chief, Operations Branch, Division of Reactor Safety (DRS), RII
C. Evans, Regional Counsel, RII
B. Uyrc, Acting Director, Enforcement and Investigation Coordination Staff (EICS), RII
L. Watson, Acting Enforcement Specialist, EICS, RII
D. Seymour, Project Engineer, DRP, RII

GEORGIA POWER COMPANY

J. Woodard, Senior Vice President, Georgia Power Company (GPC)
J. Beckham, Jr., Vice President, Hatch Project
H. Sumner, General Manager, Hatch
P. Wells, Operations Manager, Hatch
S. Bethay, Licensing Services Manager, Southern Nuclear Operating Company
S. Tipps, Manager Nuclear Safety and Compliance, Hatch
S. Brunson, Senior Engineer, GPC
E. Eckert, Plant Performance Engineer, General Electric