UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-321/94-09 and 50-366/94-09

Licensee: Georgia Power Company
P.O. Box 1295
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Docket Nos.: 50-321 and 50-366 License Nos.: DPR-57 and NPF-5

Facility Name: Hatch Nuclear Plant

Inspection Conducted: March 21, 1994 - March 28, 1994

Inspectors:	MW Sinkule for Leonard D. Wert, Jr., Sr. Resident Inspector	4/8/94 Date Signed
	MW Sinkule for Edward F. Christnot, Resident Inspector	4/8/94 Date Signed
	MW Sinkule for Bobby L. Holbrook, Resident Inspector	4/8/94 Date Signed
Approved by:	Pierce H. Skinner Pierce H. Skinner, Chief, Project Section 3B Division of Reactor Projects	4/8/94 Date Signed

SUMMARY

Scope: This special inspection reviewed a loss of shutdown cooling flow incident which occurred at Hatch Unit 2 on March 17, 1994. In addition to close observation of the licensee's investigation, the inspectors observed some of the recover, actions, held numerous discussions with involved personnel, performed independent reviews of records and analyses of the transient, and assessed the safety and regulatory issues involved.

Results: Two apparent violations were identified:

The incident involved an inadvertent loss of shutdown cooling flow which resulted in a significant increase in reactor coolant temperatures and reactor pressurization. The plant approached conditions in which containment was required without primary containment integrity. The initiating cause of the interruption of forced reactor cooling fl was very similar to a loss of shutdown cooling flow at Hatch Un 1, which occurred on April 14, 1993.

Significant procedural weaknesses related to high decay heat conditions were identified. Procedures stated that if reactor water level was maintained greater than 53 inches, without forced flow, then adequate natural circulation was ensured and the monitored temperature indications would reflect the conditions in the core

region. Information from the review of this event indicated that this was not accurate. Procedures for monitoring reactor conditions were not adequate to ensure that Operational Conditions were not inadvertently changed. The procedure for loss of shutdown cooling did not contain adequate directions regarding the difference between monitored temperatures and core temperatures during certain conditions. The guidance did not ensure adequate monitoring for reactor pressurization. The procedure for restoration of shutdown cooling systems or containment integrity did not sufficiently address exigent circumstances (paragraph 7).

Under the high decay heat condition during the event, monitoring was not required at frequent enough intervals to ensure that a loss of shutdown cooling would be identified with sufficient time to place other cooling systems in operation before reactor temperatures approached boiling. Increased monitoring commensurate with decay heat loading was to be implemented after a loss of shutdown cooling event that occurred on April 14, 1993 (paragraph 7). Several other corrective actions from the previous event were effective and improved the timeliness in recognition of the loss of shutdown cooling flow [illegible] this event (Paragraph 8).

The inspec[illegible]o concluded that the response of control room personnel [illegible]vent was timely and in accordance with the approved pr[illegible]es.

The inspectors concluded that the licensee did not demonstrate appropriate sensitivity regarding shutdown cooling system vulnerabilities, particularly during high decay heat conditions. The licensee's plans were to render the "A" train of RHR inoperable only several hours after the time that this event occurred. Although these actions were permissible by TS, more conservative planning was appropriate, especially given the licensee's previous incidents involving similar losses of shutdown cooling.

The licensee's event review team performed a detailed review of the event. This team's efforts initially identified that the reactor had become pressurized during the incident.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. Betsill, Unit 2 Operations Superintendent
*C. Coggin, Training and Emergency Preparedness Manager
*D. Davis, Plant Administration Manager
*W. Eason, Southern Nuclear Company, SAER
*R. Godby, Maintenance Superintendent
*G. Goode, Engineering Support Manager
*M. Googe, Outages and Planning Manager
*J. Hammonds, Regulatory Compliance Supervisor
*W. Kirkey, Health Physics and Chemistry Manager
*M. McLead, Plant Operator
*C. Moore, Assistant General Manager - Operations
*D. Read, Assistant General Manager - Plant Support
*K. Robuck, Manager, Modifications and Maintenance Support
*A. Singer, Plant Equipment Operator
*L. Sumner, General Manager - Nuclear Plant
*S. Tipps, Nuclear Safety and Compliance Manager
*P. Wells, Operations Manager

Other individuals:

*E. Hummel, General Electric engineer

Other licensee employees contacted included technicians, operators, mechanics, security force members and staff personnel.

NRC Resident Inspectors

*L. Wert
*E. Christnot
*B. Holbrook

* Attended exit interview

Acronyms and abbreviations used throughout this report are listed in the last paragraph.

2. Description of Incident (71707)

On March 17, 1994 at approximately 11:31 a.m. an interruption in shutdown cooling flow occurred on Unit 2. The reactor had been manually scrammed about 30 hours prior to the incident and was in cold shutdown as planned for a refueling outage. Unit 2 primary and secondary containment integrity had been breached as permitted by TS. The RCS was intact and the reactor vessel head was tensioned with the two head vent valves open. The "B" RHR loop was operating in SDC. Both CS systems, the SRVs and the "A" RHR loop were available.

The loss of SDC flow was initiated by a CR panel fuse blowing which caused the "B" SDC injection valve (2E11-F015B) to shut. The fuse was apparently blown when an engineer lifted a wire bundle located in a CR back panel to confirm a wire number. At the time of this report the licensee was still offloading fuel and a closer examination of the panel was not prudent.

The engineer saw an arc or spark when he lifted the bundle and promptly informed the shift supervisor. A control board walkdown identified that the F015B RHR B loop injection valve, was shut about 9 minutes after the fuse blew. Attempts to reopen the valve were not successful due to a partial Group II isolation signal present. The abnormal procedure for the loss of SDC was entered and reactor water level was increased as required. About 20 minutes after the shut valve was identified, the SS directed that the A loop of RHR be placed in SDC. At about 12:55 p.m., SDC was restored. Indicated temperatures (RWCU inlet) increased from about 168 degrees F to 195 degree F during the incident. The licensee's subsequent investigation identified that the reactor had pressurized to about 8 psig. Calculations by GE indicated that RCS average or bulk temperature may have reached as high as 210 degree F. No equipment failure occurred during the recovery and operators actions were timely and appropriate.

3. Sequence of Events: (71707)

1:49 a.m.	(3/16/94) Hatch Unit 2 was manually scrammed as planned to commence the scheduled refueling outage.
11:41 p.m.	(3/16/94) The B loop of RHR was placed in shutdown cooling. Both reactor recirculation pumps were secured.
2:54 a.m.	(3/17/94) Unit 2 reached cold shutdown.
4:05 a.m.	Unit 2 drywell head manway cover removed. Primary containment was breached. Unit 2 drywell became part of Unit 1 secondary containment. Secondary containment test completed satisfactorily.
6:15 a.m.	Unit 2 drywell head lifted and moved to designated location.
9:19 a.m.	Unit 2 reactor building doors opened. Equipment hatches were still in place, DW airlock was operable for DW access. Reactor vessel head remained tensioned.
10:54 a.m.	34GO-OPS-015-2S: Maintaining Cold Shutdown, hourly control room panel checks, which include SDC flow verification was completed satisfactorily.
11:15 a.m.	A Southern Nuclear engineer began to verify wiring in CR back panel 2H11-P623. The verification was in preparation

for replacement of several CR 120 relays. 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 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 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 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-F022 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.

2:00 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.

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.

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 p.m. (approximately 1 hour 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 ¼ 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 of the NRC recommendations resulting from TMI-2 lessons learned. An NRC item from NUREG-0578 required that design information be provided to demonstrate that non-condensible gases can be vented from the primary coolant system. The letter stated that the venting capability, including the vessel head vent, was described in the FSAR and this was sufficient to satisfy the category "A" requirements for the item.

The System Evaluation Document for the nuclear boiler system (B21) also supports the position that the head vent path is to permit venting during filling for hydrostatic tests and venting of noncondensibles in the vessel head space during cooldown. It also stated that the vent line is connected to one of the main steam lines to remove noncondensible disassociated gases which accumulate in the vessel head space during operation.

From these reviews, and information that the ERT obtained from GE, the inspectors concluded that the vent lines were intended to vent non-condensables from the head space, not prevent pressurization by steam formation during a substantial heatup of reactor coolant during a loss of a SDC event.

A review of P&ID 26001 indicated that some modifications had been performed on the reactor head vent piping. Some piping sections had been apparently removed and capped off. DCR 88-132 had removed and capped a section of 2 inch diameter piping containing two manual valves which had been a path parallel to the 1/2 inch diameter piping and vent valves. The inspectors reviewed this design change and the associated safety evaluation. The evaluation stated that the manual valves were normally locked closed and had been installed only to increase venting capability for hydrostatic testing. The manual valves were considered as "redundant to the air operated valves" and had apparently required repairs for leaks.

The inspectors questioned if the configuration of Hatch's vent path was typical of most BWRs. It was determined that many other BWRs still have two inch vent paths installed. GE was requested by the licensee to calculate the pressure which would have been reached had a two inch path existed at Hatch during the loss of SDC event.

The inspectors concluded that the licensee's removal of the two inch vent path piping in 1989 could be considered to be not prudent, given certain potential heatup and pressurization scenarios. However, the removal was not in violation of regulatory requirements. The licensee's ERT proposed that reinstallation of the two inch line be evaluated. Additionally, training and procedures should more accurately reflect the capabilities of the reactor head vent line.

7. Procedural Issues (71707) (62703)

During the review of this event, several procedural deficiencies were noted. While the existing procedures did result in successful restoration of shutdown cooling flow in this instance, the inspectors 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 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 procedure does not contain guidance on how the metal temperatures should be used.

Step 4.7 of procedure 34AB-E11-001-2S, described the reactor vessel pressure indications to monitor at 15 minute intervals. The procedure stated that instruments 2C32-R605A, B, and C or the process computer may be used. The indications are 0-1200 psig on a CR panel which generally will not indicate small reactor pressures which could be significant under these plant conditions. Use of these indications, along with a caution which implied that with high water level and natural circulation flow would prevent vessel pressurization if vented, resulted in operators not recognizing that the reactor had pressurized to about 8 psig during the event.

The inspectors concluded that the procedures did not adequately address high decay heat conditions. The deficiencies are identified as an apparent Violation 50-366/94-09-01: Shutdown Cooling Procedure Inadequate for High Decay Heat Conditions. The inspectors noted that Service Information Letter (SIL) 357: Control of Reactor Vessel Temperature/Pressure During Shutdown, contained recommendations to help prevent BWRs from exceeding 212 degree F while in cold shutdown with some ECCS and containment systems inoperable. The SIL discusses thermal stratification upon loss of forced cooling and the potential for pressurization and a TS violation if containment is not intact. The SIL stated that raising the reactor water level to above the pre-dryer section of the steam separator would allow natural circulation between the inside and outside of the shroud. This circulation will slow the temperature rise to 212 degrees F (and pressurization) as well as give a more accurate vessel temperature indication in the recirculation system. The SIL stated that without sufficient circulation the RWCU and recirculation system termocouples will register significantly lower than the temperature at the surface of the water. If insufficient heat is removed, this condition will result in raising the temperature at the water surface above 212 degrees F with resultant pressurization of the RPV and/or steam generation if the vessel has a venting path open. The recommendations of the SIL included not removing the SDC/RHR system from service or restricting its performance unless the reactor vessel cavity is flooded with the vessel head removal. The SIL also recommended that if the SDC/RHR system is not available and RPV head removal is not feasible then vessel metal surface temperature and pressure should be monitored and RPV water level raised, head spray operated as available, and RWCU system operated. The SIL also recommended that if the monitored temperatures and pressures are not being maintained then ensure the level is raised and prepare plant for operation above 212 degrees F.

During this event jet pump differential pressure indications were interpreted to be indicative of natural circulation in the core. Apparently, the natural circulation flowrate was not as large as the differential pressures indicated. Since the indicators are calibrated for normal operating temperatures water density differences could be causing flow to be lower than indicated.

The licensee and GE was currently evaluating the information regarding natural circulation and temperature indications gained as a result of this incident to ensure the above recommendations remain appropriate.

Step 4.3 of procedure 34AB-E11-001-2S stated that if SDC cannot be restored quickly and reactor water temperature may exceed 212 degrees F, restore primary and secondary containment and ensure systems required for entry into condition 3 are operable. In this case, because personnel were not aware of actual plant conditions, no actions were initiated to restore containment despite reactor temperatures in excess of 200 degrees F. Additionally, the inspectors noted that there are no contingency plans to restore containment quickly.

Attachment 1, to procedure 34AB-E11-001-2S, is a graph of vessel boil off time versus number of days since last critical. The inspectors noted that based on the plant conditions at the time of the loss of SDC flow, the graph indicates that boiling will occur about 1 hour after a loss of SDC. The inspectors noted that in this case, almost one hour was required for restoration of a SDC loop. The inspectors concluded that the procedural guidance for restoration of a SDC loop under exigent conditions should be improved. Additionally, the inspectors noted that the graph is difficult to utilize since initial temperature, water level, and shutdown power levels are not indicated on the graph. For times shortly after shutdown the graph is difficult to use as a reference.

During the review of the procedures and the required timeliness of actions an additional concern was noted. Procedure 34GO-OPS-015-2S: Monitoring Cold Shutdown and Refuel Parameters, required monitoring of reactor coolant hourly. Prior to the fuse blowing, the temperature was checked at 10:54 a.m. If the engineer had not informed the operators of the problem, the next check would have been performed as late as 11:45 a.m. The inspectors concluded that a 20 minute delay, under the high decay heat conditions present, could have resulted in reactor temperature in excess of 212 degree F. In response to a loss of shutdown cooling event in April 1993, this procedure was to be revised to ensure that checks were performed commensurate with decay heat loads. The inspectors concluded that the procedure was not revised appropriately since under high decay heat loads, checks should be move frequent. This is addressed as an apparent Violation 50-366/94-09-02: Cold Shutdown Monitoring Procedure Not Corrected to Frequency Commensurate with Decay Heat Load. As discussed in paragraph 8 of this report, other corrective actions for the previous incident were effective.

Section 7.4 of Procedure 52GM-MME-015-2S: Reactor Vessel Disassembly provides guidance for removal of the RPV piping. The inspectors noted that step 7.4.3 required workers to "confirm with the CR that the reactor is vented, valves 2B21-F003 and F004 are OPEN. Based on this event, the reactor may be pressurized even with the valves open. When workers unbolted a flange on the head, about one hour after SDC flow had been 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-F022 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 modification work on a control room panel (front panel) manipulation of some indication wiring in the panel 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.

The inspectors confirmed that if the incident had not occurred, the "B" 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.

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

ERT - Event Review Team
ESF - Engineered Safety Feature
F - Fahrenheit
FSAR - Final Safety Analysis Report
GE - General Electric Company
GL - Generic Letter
GPC - Georgia Power Company
HX - Heat Exchange
IN - Information Notice
IR - Inspection Report
LCO - Limiting Condition for Operation
LER - Licensee Event Report
LLRT - Local Leakrate Test
MCPR - Minimum Critical Power Ration
MSIV - Main Steam Isolation Valve
NRC - Nuclear Regulatory Commission
NRR - Nuclear Reactor Regulation
NSAC - Nuclear Safety and Compliance
PCIS - Primary Containment Isolation System
P&ID - Piping and Instrumentation Drawing
PSIG - Pounds Per Square Inch
RB - Reactor Building
RCS - Reactor Coolant System
RG - Regulatory Guide
RHR - Residual Heat Removal
RPV - Reactor Pressure Vessel
RTP - Rated Thermal Power
RWCU - Reactor Water Cleanup
SCS - Southern Company Services
SDC - Shutdown Cooling
SFP - Spent Fuel Pool
SIL - Service Information Letter
SOS - Superintendent of Shift (Operations)
SPDS - Safety Parameter Display System
SRO - Senior Reactor Operator
SRV - Safety Relief Valve
SS - Shift Supervisor
TMI - Three Mile Island
TS - Technical Specifications