September 3, 2004

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION
NRC SPECIAL INSPECTION REPORT 05000461/2004007

Dear Mr. Crane:

On July 26, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed a special team inspection at your Clinton Power Plant. The enclosed report documents the inspection findings which were discussed with Mr. R. Bement and other members of your staff on July 28, 2004.

On July 13, 2004, an automatic shutdown occurred from 95.3 percent power due to an apparent instantaneous neutral phase over-current fault trip on the main power transformer. All plant systems operated normally on the automatic shutdown with the exception of the “A” recirculation pump that failed to start in slow speed (15 Hertz). In addition, during the recovery, the “B” feedwater pump tripped on low suction pressure when operators were securing the “A” feedwater pump. On July 14, 2004, further complications occurred when the reactor pressure vessel water level dropped about 24 inches, resulting in another reactor protection system actuation.

Using the deterministic criteria provided in Management Directive 8.3 and Inspection Procedure 71153, “Event Followup,” a special inspection was initiated. Specifically, the unexpected system interactions that resulted in the loss of coolant inventory, and the multiple electrical equipment failures could indicate concerns with operational performance.

This inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission’s rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel involved with these events.

Based on the results of this inspection, one NRC-identified finding and one self-revealed finding of very low safety significance (Green), both of which involved violations of NRC requirements, were identified. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these two findings as Non-Cited Violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory

Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, US Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspectors Office at Clinton Power Station facility.

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Sincerely,

/RA by Patrick L. Hiland acting for/

Steven A. Reynolds, Acting Director
Division of Reactor Projects

Docket No. 50-461
License No. NPF-62

Enclosure: Inspection Report No. 05000461/2004007
w/Attachment: Supplemental Information
Special Inspection Charter

cc w/encl: Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2004007
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Route 54 West Clinton, IL 61727
Dates:	July 15 through July 21; and July 26, 2004
Inspectors:	C. Brown, Resident Inspector, Team Leader R. Walton, Operations Engineer
Observer:	Daneira Melendez-Colon, Nuclear Safety Professional
Approved by:	Ann Marie Stone, Chief Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000461/2004007, 07/15/2004 - 07/26/2004; Clinton Power Station. Special Inspection for Clinton Power Station July 13, 2004 reactor automatic shutdown and July 14 level transient.

This special inspection examined the facts and circumstances surrounding an automatic shutdown following a turbine load reject during a severe thunderstorm on July 13, 2004, and a sudden reduction of reactor vessel inventory on July 14, 2004. The inspection was conducted by the resident inspector and a Region III inspector in accordance with Inspection Procedure 93812. This inspection identified two Green findings with two associated Non-Cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “Green” or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. <u>Inspector-Identified and Self Revealing Findings</u>

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance, with an associated Non-Cited Violation, was identified by the inspectors. Specifically, the licensee failed to analyze how a feedwater pump modification affected the operators’ duties after an automatic shutdown. As a result of the modification, operators should have been directed, by procedure and training, to trip the “B” feedwater pump following an automatic shutdown. One of the causes of this finding related to the cross-cutting area of problem identification and resolution, in that, the licensee did not identify the discrepant procedure or training during investigation of a previous event.

  The issue was more than minor because if left uncorrected, it could be reasonably viewed as a precursor to a significant event. Specifically, it caused unnecessary complications to the automatic shutdown sequence, placed extra importance on the motor-driven reactor feedwater (MDRF) pump and could challenge the high-pressure emergency core cooling systems (ECCS) during a motor-driven feedwater pump outage. The inspectors determined that the finding could not be evaluated in accordance with IMC 0609, “Significance Determination Process.” Therefore, this finding was reviewed by the Regional Branch Chief in accordance with IMC 0612, Section 05.04c, and determined to be of very low safety significance because the MDRF pump did start and the high pressure ECCS systems were operable. The finding was assigned to the mitigating system cornerstone. The issue was a Non-Cited Violation of Criterion II of 10 CFR 50 Appendix B. The licensee took immediate corrective action to revise the procedure, installed a robust barrier over the “A” feedwater pump control switch, and briefed all operators on the effects of the modification. (Section 02.1.b.3)

- Green. A finding of very low safety significance, with an associated Non-Cited Violation, was self-revealed. Specifically, Clinton Power Station Procedure 3312.03, “Shutdown Cooling and Fuel Pool Cooling and Assist,” was inadequate because it allowed the operators to create voids inside system piping while preparing to place the “B” residual heat removal (RHR) system in the shutdown cooling mode of operation. When sufficient differential pressure developed to open the RHR pump discharge check valve, about 2000 gallons of water unexpectedly drained from the reactor pressure vessel into the RHR system and produced a reactor automatic shutdown signal and Level 3 isolation on low reactor water level. The “B” RHR system was subsequently declared inoperable.

The finding was more than minor because it affected the Reactor Safety/Mitigating System Cornerstone and if left uncorrected, it would become a more significant safety concern. Specifically, voided piping could produce a system water hammer when the residual heat removal water pump is started in shutdown cooling mode and render the system inoperable. The finding was determined to be of very low safety significance because there was no design deficiency, no actual loss of safety function, no single train loss of safety function for greater than the Technical Specification allowed outage time and no risk due to external events. The licensee revised the shutdown cooling steps in the procedure, briefed all operators on the apparent cause, and entered the event into its corrective action system. The issue was a Non-Cited Violation of Criterion V of 10 CFR 50 Appendix B. (Section 02.2b.1).

B. **<u>Licensee-Identified Violations</u>**

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Events

Synopsis of Events

At 4:08 p.m., during a severe thunderstorm on July 13, 2004, an automatic shutdown occurred from 95.3 percent power (which is the maximum power achievable under the authorized power up rate) due to an apparent instantaneous neutral phase over-current fault trip on the main power transformer. All plant systems operated normally on the automatic shutdown with the exception that the “A” recirculation pump ran back per design but failed to restart on low speed (15 Hertz). In addition, during the recovery, the “B” feedwater pump apparently tripped when operators secured the “A” feedwater pump.

Further complications occurred at 12:45 a.m., on July 14, 2004, when the reactor pressure vessel level dropped about 24 inches resulting in another reactor protection system and isolation actuation. Before the second event, the unit was in Mode 3 [hot standby] with reactor pressure at 18 psig and level stable at 32 inches. The licensee was preparing the “B” residual heat removal (RHR) system for the shutdown cooling mode of operation by warming the heat exchanger portion of the system through natural circulation using a flow path from the reactor vessel to the radioactive waste system. Once secured, the licensee verified the system was filled and vented; however, pressure and temperature in the heat exchanger decreased. When the licensee re-established the warming sequence, reactor vessel level dropped. The licensee believed that the “B” RHR discharge check valve (1E12-F031B) was stuck off its seat which caused water to drain from the system. Two additional attempts to prepare the “B” RHR system for shutdown cooling were made; however, when similar indications of a loss of temperature and pressure in the discharge piping occurred, further attempts to prepare the system were stopped.

Inspection Scope

Based on the risk and deterministic criteria specified in Management Directive 8.3, “NRC Incident Investigation Program,” and Inspection Procedure 71153, “Event Followup,” and due to the equipment performance problems which occurred, a Special Inspection was initiated in accordance with Inspection Procedure 93812, “Special Inspection.” Specifically, the unexpected system interactions that resulted in the loss of coolant inventory, and the multiple electrical equipment failures, could indicate concerns with the licensee’s operational performance.

The purpose of the inspection was to evaluate the facts and circumstances surrounding the events as well as the actions taken by licensee personnel in response to the unexpected system performance issues encountered. The inspection charter is attached.

In particular, the inspection focused on the adequacy of the licensee’s evaluation of the July 13, 2004, reactor automatic shutdown including: (1) the cause of the turbine-generator trip which resulted in the reactor shutdown, (2) the unexpected response of the Gas Circuit Breaker 4510, (3) the evaluation of the effects of the automatic shutdown including associated potential damage to the main power transformer and switchyard equipment, (4) the cause of the “A” recirculation pump failing to shift to low speed as designed, and (5) the cause of the “B” feedwater pump trip including an evaluation of effectiveness of a modification installed in February 2004 to prevent such trips.

The inspection also focused on the adequacy of the licensee's evaluation of the July 14, 2004, loss of reactor water level event including: (1) an assessment of operators' performance during the preparation to place "B" RHR system in the shutdown cooling mode of operation, (2) an assessment of the procedures used to place "B" RHR system in the shutdown cooling mode of operation including the decision to proceed following the initial temperature and pressure anomalies, (3) the cause for the unexpected lowering of reactor water inventory and subsequent inability to place the system in the shutdown cooling mode, and (4) the unexpected performance of the "B" RHR discharge check valve, the RHR heat exchanger service water relief valve, and other equipment concerns identified during the course of the licensee's investigation.

1 REACTOR SAFETY

Cornerstones: Initiating Events and Mitigating Systems

01 Sequence of Events (93812)

a. Inspection Scope

The inspectors reviewed logs, alarm printouts, and other documentation; interviewed licensee personnel; and developed the following sequence of events for the July 13, 2004, automatic shutdown and the July 14, 2004, loss of reactor water level event.

July 13, 2004, Automatic shutdown

Day	Time	Event Description
7/13	16:08:40	The unit was in Mode 1 with reactor power at 95.3 percent and the main generator gross output at 1092 MWe. A severe thunderstorm was in progress outside the plant. A fault occurred on one of the off-site supply lines to the switchyard ring bus (the Brokaw line) causing 345kV gas circuit breakers (GCBs) 4502 and 4506 to open. The main power transformer (MPT) received an instantaneous neutral phase over-current trip. This resulted in a main generator trip system 1 lockout that tripped open GCB 4510 with a subsequent reactor automatic shutdown. [The fault was later determined to be from a close-in lightning strike creating a 10,000 ampere fault to ground at a grid substation.]
	16:08:42	Following the reactor automatic shutdown, reactor pressure vessel (RPV) level dropped to Level 3 (8.9 inches) as expected. The main control room (MCR) operators entered Emergency Operating Procedure (EOP-1), RPV Control.
	16:11:31	The "B" turbine driven reactor feedwater (TDRF) pump tripped due to low suction pressure after a 6 second time delay.

	16:11:34	The reactor operator tripped TDRF pump "A" according to immediate actions of the Reactor SCRAM Off-Normal procedure. With both TDRF pumps tripped, the "C" motor driven feedwater pump started automatically, as designed. [At the time of the event, it appeared that the operator tripped the "A" TDRF pump prior to the "B" pump automatic trip.]
	16:17:00	Reactor recirculation (RR) "B" pump down shifted to slow speed. However, the RR "A" pump started to downshift to slow speed but failed to start in slow speed and tripped off. The reactor operator noted the failure to start and entered the Abnormal Reactor Coolant Flow procedure.
	16:20:00	The reactor operator shut RR "A" pump discharge valve per the instructions of the Abnormal Reactor Coolant Flow procedure.
	17:09:00	The operators reset the Reactor automatic shutdown signal.
	18:07:00	The licensee notified the NRC per 10 CFR 50.72 for the reactor protection system actuation.
	20:13:00	Shift Manager contacted Illinois Power to have the switchyard walked down looking for any faults or damage.

<u>July 14, 2004, Unexpected Reduction of Reactor Water Level</u>

7/13	20:45:00	The control room operators closed the RHR "B" suppression pool suction valve, 1E12-F004B, in preparation for flushing the shutdown cooling (SDC) flow path.
	21:00	RPV pressure is 140 psig with Pressure Set at 142 psig.
	22:37:00	Per CPS 3312.03, "RHR-Shutdown Cooling & [and] Fuel Pool Cooling and Assist," operators opened the shutdown cooling outboard isolation valve, 1E12-F009, and the shutdown cooling inboard isolation, 1E12-F008, for warmup of RHR "B" shutdown cooling loop.
	~23:09	The oncoming reactor operator commenced lowering reactor pressure with the control room supervisors concurrence but without discussing the action with the operators warming up RHR "B". [Prior to this, reactor pressure was held constant.]
7/14	00:17:00	Control room operators observed a drop in pressure in RHR "B" after securing from warming and a decision is made to re-perform a fill and vent of the RHR "B" heat exchanger with cycled condensate water. The operators observed a drop in the indicated temperature and secured from the fill and vent process.

00:30:00 The operators re-entered RHR “B” system warm up. In accordance with procedure, operators throttled open RHR second isolation to radwaste valve, 1E12-F040, for approximately 8 to 9 seconds and throttled open RHR “B” heat exchanger outlet valve, 1E12-F003B, for 1 to 2 seconds.

00:43:00 Reactor pressure was about 18 psig and vessel level was 32 inches. The operators throttled open E12-F040 an additional 3 to 4 seconds to continue the warming process. Almost immediately RPV level dropped 23 inches in approximately 20 seconds.

The operators received a reactor automatic shutdown actuation and a containment isolation signal due to the reactor water level dropping below Level 3. The level decrease stopped when the 1E12-F008 and 1E12-F009 valves closed on the containment isolation signal.

The operators appropriately re-entered EOP-1 and the Reactor SCRAM Off-Normal procedure. The lowest RPV level observed was +9.0 inches in the narrow range. [This is about 14 feet above the top of active fuel.] The operators immediately recovered vessel level using the condensate system.

02:55:00 The licensee notified the NRC per 10 CFR 50.72 for the second RPS and containment isolation valve actuation.

04:00:00 Initial assessment showed a void was drawn in RHR “B” heat exchanger and piping. The operators believed that the discharge check valve had not shut completely during the RHR “B” flush and had subsequently rapidly opened causing excessive flow into the RHR “B” system causing RPV level to lower. Issue Report 235832 was generated to document the event. Operators commenced raising reactor pressure to approximately 60 psig to facilitate placing “A” RHR train in the SDC mode. Because of the potential problem with RHR “B” discharge check valve, RHR “B” was declared inoperable and unavailable.

05:36:00 Operators started and stopped the RHR “B” pump in pool-to-pool operation per CPS 3312.01, “Residual Heat Removal (RHR),” to ensure the discharge check valve was seated correctly when the pump was secured.

10:31:00 The operators commenced another attempt to place RHR “B” in the SDC mode per CPS 3312.03; however, this effort was stopped due to potential RHR “B” to shutdown service water leakage. Operators determined leakage was from a shutdown service water relief valve (1SX208B).

12:45:00 The licensee decided to continue with CPS 3312.03 for placing RHR “B” in the SDC mode.

	13:45:00	The operators determined that RHR “B” system cannot maintain pressure and temperature requirements to be placed in the SDC mode [The reason(s) were not understood at that time but pressure and temperature would not stay in specification after warm up].
	16:35:00	Operators vented RHR “B” per CPS 3312.01 with cycled condensate pressure via the 1E12-F063B valve. Solid streams of water were obtained with no presence of air and RHR heat exchanger “B” pressure was ~110 psig. Once the cycled condensate valve was shut, pressure slowly decayed to ~50 psig, before the RHR “B” pump suction valve (1E12-F004B) was opened. The pressure continued to decay below the low pressure alarm point and the alarm annunciator was received again.
	17:49:00	The operators opened and then shut the RHR “B” heat exchanger vent valves, 1E12-F357B and 1E12-F358B. The pressure in RHR “B” went from ~18psig on the computer point to ~8psig during this evolution with a solid stream of water and no air present -- once the valves were shut, pressure in the system slowly recovered to ~22 psig.
	17:58:00	Due to problems encountered with restoring RHR “B” in accordance with CPS 3312.03, a decision was made to begin RHR “A” shutdown cooling flushing/warmup preparations.
7/15	06:06:10	Started RHR “B” pump in pool-to-pool operation, for troubleshooting on system pressure decay rate. The pump was stopped at 06:17 after operating satisfactorily.
	11:50:00	RHR “B” was filled and vented, and the post maintenance test was satisfactory. The licensee declared RHR “B” operable and available for all functions except the SDC mode.
7/16	10:23:00	Between this time and 11:38, the operators successfully performed the water leg pump operability surveillance testing.
7/17		During the day, the licensee performed inspections on the discharge check valve, 1E12-F031B. No problems were noted. Post maintenance test was satisfactory.
	14:22:00	Based on inspections and fill/vent operation, the licensee declared the RHR “B” system operable for all modes of operation. Procedures to prevent void conditions were revised.

The inspectors determined that the operators accumulated an additional 10 millirem (mrem) of radiation exposure while attempting to place “B” loop in the SDC mode. Operators placing “A” loop into shutdown cooling received 14 mrem of exposure. There were no radiological consequences to this event.

The inspectors reviewed the licensee’s event classifications and notifications. No concerns were identified. The licensee properly classified and reported the events.

b. Findings

No findings of significance were identified.

02 Adequacy of Licensee Evaluation of Events and Corrective Actions (93812)

02.1 Automatic Reactor Shutdown on Instantaneous Neutral Phase Over Current

a. Inspection Scope

On July 13, 2004, with the plant in CPS 4302.01, “Tornado/High Winds,” due to a tornado warning and with a severe thunderstorm in progress, a fault occurred when lightning struck one of the off-site supplies to the switchyard ring bus (the Brokaw line). Gas Circuit Breakers (GCBs) 4502 and 4506 opened to isolate the fault from the generator and the ring bus. The main power transformer (MPT) received an instantaneous neutral phase over-current trip during the fault which resulted in a main generator trip system 1 lockout actuation. The lockout tripped open GCB 4510, leading to the subsequent reactor automatic shutdown at 4:08 p.m. Before the event, the unit was operating at 95.3 percent reactor power. Subsequent to the automatic shutdown, the “B” TDRF pump appeared to trip when an operator tripped the “A” TDRF pump per the automatic shutdown procedure. Additionally, the “A” reactor recirculation pump failed to shift to slow speed on the loss of the TDRF pumps.

The inspectors assessed the licensee's investigation report for this event, and interviewed control room operators and engineering personnel. Additionally, the inspectors reviewed system drawings, control room logs, maintenance records, recorded data traces, and operating procedures.

b. Observations and Findings

.1 Unexpected opening of the 4510 generator output breaker

The licensee's troubleshooting team identified that the apparent cause of GCB 4502 and 4506 opening was a lightning strike on the “A” phase of the Brokaw line approximately 4.5 miles from the station. The GCBs 4502 and 4506 opened as designed to isolate the fault (about 10,000 amps). The fault was cleared in 2.5 cycles when the GCBs 4502 and 4506 opened. About 5400 amps of the fault current was supplied by the Clinton main generator. The main generator protection system ITH relay was set to trip at 1600 amps and actuated at about 0.7 cycles into the event. The lightning strike occurred at a point on the voltage waveform on the A phase that caused a neutral instant over-current fault signal to be generated due to a saturated current transformer. This fault signal caused the main generator trip system 1 lockout relay to energize and open (lockout) both generator output breakers. The fault signal was generated before the ring bus breakers (GCBs 4506 and 4502) could isolate the lightning strike but the actual lockout occurred after GCBs 4506 and 4502 had opened. The lockout tripped open GCB 4510 (one main generator output breaker) and prevented GCB 4506 (the other main generator breaker) from reclosing with a subsequent reactor automatic shutdown due to generator load reject. All components worked as designed and there were no instances of equipment damage or malfunction from this event. To prevent a similar occurrence in the future, the licensee installed a different (slower acting) fault sensing relay which will allow the ring bus breakers to isolate a fault caused by a close-in lightning strike before the relay energizes to lockout both generator output breakers.

The inspectors reviewed the results of the licensee's trouble-shooting efforts including comprehensive reviews of the protection schemes, inspections for equipment damage, and independent assessment of the data recordings and test results. The inspectors concurred that no equipment damage or malfunction occurred. The inspectors found the licensee's evaluation of this event to be thorough and comprehensive. No findings of significance were identified.

.2 Reactor recirculation pump "A" failed to shift to slow speed on a loss of feed water

The licensee's investigation revealed that both reactor recirculation pumps had started to downshift to slow speed when the turbine driven feedwater pumps tripped. However, the "A" reactor recirculation (RR) pump failed to start in slow speed (15 Hertz) and tripped off. The operator noted the failure to start in slow speed and shut the pump discharge valve per procedure. The licensee completed extensive investigations, including instrumenting the relays and breakers associated with the "A" recirculation pump without finding any definitive cause for the pump failing to start in slow speed. The licensee then measured the resistance across the relay and breaker contacts and found some contacts with possibly higher resistance than normal. The licensee put actions in place to clean these contacts after the RR "A" pump was shifted to fast speed during the reactor startup. After assessing the risk of leaving external monitoring instruments installed vice the risk of not getting further information if the pump failed to downshift again, the licensee removed the extra monitoring equipment.

The inspectors assessed the investigations and evaluation for the RR "A" pump failing to start in slow speed and found them to be acceptable. Although no definitive cause was found, the licensee's plans to burnish some contacts that had exhibited increased resistance after the RR "A" pump was running in fast speed were considered reasonable. No findings of significance were identified.

.3 Turbine driven reactor feed pump (TDRF pump) "B" tripped when the operator tripped "A" TDRF pump following the reactor automatic shutdown

Introduction: The inspectors identified a Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program," for failing to adequately notify operators of the impact of a modification. Specifically, the licensee modified a time delay trip for the "A" TDRF pump but did not incorporate this change in the shutdown procedures or in operator training. After a high power load-reject automatic shutdown, the operators should have been directed by procedure and training to trip the "B" TDRF pump as a result of the modification.

Description: During an automatic shutdown from high power, both feed pumps speed up in response to the rapid drop in reactor water level due to void collapse, creating low suction pressure. To alleviate this condition, the operators had been trained to trip a feed pump soon after an automatic shutdown occurred. The operators routinely tripped the "A" TDRF pump.

In December 2003, modification EC 338996, “Modify Timing of TDFW Pump Trip On Low Suction Pressure So That Only One Feed Pump Is Tripped At A Time - To Avoid SCRAM,” Revision 000, was installed. One purpose of the modification was to add an extra 6 second time delay on the "A" TDRF pump to prevent simultaneously tripping both TDRF pumps on a loss of suction pressure. As a result, the "A" pump had a 12 second delay and the "B" pump had a 6 second delay. With this condition, operators should manually trip the “B” TDRF pump first to prevent the “A” TDRF pump trip.

The inspectors reviewed the modification package and determined that the licensee had not fully considered the impact of the modification on the feedwater system, specifically, operator training and procedures. For example, the inspectors determined that the package did not reference CPS 4100.01, “Reactor Scram,” in the affected procedures section; did not assess the effects of the modification on the operators in the 10 CFR 50.59 screening evaluation; and did not assess the effects on operator training. The inspectors concluded that the operators’ training on this modification consisted of an article in the "Core Newsletter" which only described the facts of the modification. The automatic shutdown procedure which required the operators to trip a feed pump was not revised to direct operators to specifically trip the “B” TDRF pump. Additionally, interviews with operators revealed that they had still been preferentially tripping the “A” feed pump in their most recent training cycle (June 2004).

In February 2004 additional modifications to the feedwater system were made to support the extended power up rate (EPU). The modifications included resetting the low suction pressure trip from 250 psig to 325 psig and resetting the TDRF pump high speed stops to obtain over 1000 gpm additional flow. This additional flow caused the condensate booster pumps to be lower on their operating curve; therefore, producing less discharge pressure (~90 psig). These modifications significantly increased the probability of tripping a feedwater pump on low suction pressure during a high power load reject.

In addition, the inspectors determined that the licensee had previous opportunity to identify the training and procedure deficiencies. Specifically, in March 2004, an automatic shutdown from a main power transformer isophase bus duct fault occurred. At that time, the NRC inspectors questioned the licensee’s implementation of the reactor water level set point/set down feature (upon a reactor shutdown, the reactor level setpoint is raised then lowered as compared to other sites where the level is lowered). During their investigation, the licensee ran a number of simulator runs but did not recognize that the operators continued to trip the “A” TDRF pump first; therefore, did not identify the need to revise plant procedures or training. The licensee is still evaluating the effects of the set point/set down feature with respect to the TDRF pump modifications.

<u>Analysis</u>: The inspectors determined that not recognizing or analyzing the effects of a modification (EC 338996) sufficiently to ensure that the operators did not inadvertently increase the plant challenges after an automatic shutdown was a performance deficiency warranting a significance determination. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC)

0612, “Power Reactor Inspections Reports,” Appendix B, “Issue Disposition Screening,” issued on June 20, 2003. The issue was more than minor because if left uncorrected, it could be reasonably viewed as a precursor to a significant event. Specifically, the near simultaneous tripping of both TDRF pumps complicated the automatic shutdown sequence and placed more importance on the MDRF pump to automatically start and supply feedwater to the reactor. Additionally, the high-pressure ECCS systems could be challenged during a motor-driven feedwater pump outage.

The inspectors determined that this deficiency affected the cross-cutting area of Problem Identification and Resolution. Problem Identification and Resolution was affected because the effects of the modifications on the operators performance of their duties were not identified when preparing the modification package or during the licensee investigation and simulator runs after the March 2004 automatic shutdown.

The inspectors determined that the finding could not be evaluated in accordance with IMC 0609, “Significance Determination Process.” Therefore, this finding was reviewed by the Regional Branch Chief in accordance with IMC 0612, Section 05.04c, and determined to be of very low safety significance (Green) because the MDRF pump did start and the high pressure ECCS systems were operable. The finding was assigned to the mitigating system cornerstone.

Enforcement: Title 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program," requires, in part, that the Quality Assurance program shall be documented by written policies, procedures, or instructions and shall be carried out in accordance with these policies, procedures, or instructions. Paragraph 2.6 of Chapter 3, “Design Control,” of the Clinton Quality Assurance Topical Report requires that plant personnel be made aware of design changes or modifications which may affect the performance of their duties. Contrary to the above, from December 2003 to July 2004, the licensee failed to adequately train operators and revise procedures resulting from modification EC 338996, “Modify Timing of TDFW Pump Trip On Low Suction Pressure So That Only One Feed Pump Is Tripped At A Time - To Avoid SCRAM,” Revision 000. Specifically, the licensee failed to identify that the plant automatic shutdown response procedure needed revision and that the reactor operators were not made aware to preferentially trip the “B” turbine-driven feedwater pump vice the “A” pump after an automatic shutdown from high power. The licensee took immediate corrective action to revise the automatic shutdown procedure, installed a robust barrier over the “A” TDRF pump control switch, and briefed all operators on the effects of the modification.

This was a violation of Criterion II of 10 CFR 50, Appendix B. However, because this violation was of very low risk significance and was captured in the licensee's corrective action program (CR 237898), this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2004007-01).

02.2 Loss of Reactor Vessel Inventory When Placing “B” Loop into Shutdown Cooling (93812)

a. Inspection Scope

After the automatic shutdown on July 13, 2004, operators attempted to place the “B” RHR loop in the SDC mode of operation using CPS 3312.03, “Shutdown Cooling and Fuel Pool Cooling and Assist” procedure. The first section of this procedure involved flushing cycled condensate system water through the SDC piping to remove any impurities from the system. After flushing was completed, the procedure had operators drain hot low pressure water from the RPV to warmup the pump and piping up to the heat exchanger. After the inlet to the heat exchanger was warmed to within 50°F of RPV temperature, the RHR pump would be started to place the system in the SDC mode.

Control room personnel reduced RPV pressure in parallel with flushing operations then held RPV pressure between 60 psig to 75 psig for initial system warmup. Around 11:30 p.m. on July 13, new control room operators commenced reducing reactor pressure. The need to hold reactor pressure steady during initial system warmup was not discussed during the turnover. Operators assigned to warmup SDC piping had reached the desired temperature and repositioned the heat exchanger outlet and bypass valves at about 12:13 a.m. when parameter displays recorded a low pressure spike in the system. Operators filled and vented the SDC system causing heat exchanger inlet temperature to drop when cooled by the cycled condensate water. Control room operators, believing that the piping had cooled down too far, consulted with, and received permission from the control room supervisor to reenter the SDC piping warmup procedure. Operators then reestablished the warmup flow path from the RPV by draining water from the SDC system to the radwaste system. This condition existed for about 10 minutes when the flow rate was unexpectedly increased. Trend recorders showed a large temperature increase on the top of the heat exchanger as 2000 gallons of RPV water entered into the SDC system. A containment isolation signal (Level 3) ceased flow from the RPV into the SDC system. The condensate system immediately refilled of the RPV to its normal level.

The inspectors assessed the licensee’s investigation report for this event, and interviewed control room operators and engineering personnel. Additionally, the inspectors reviewed system drawings, control room logs, radiation exposure records, maintenance records and operating procedures. The inspectors also walked down affected portions of “B” RHR piping.

b. Observations and Findings

.1 Assessment of Shutdown Cooling Procedures and Cause of the Loss of Inventory

Introduction: A Non-Cited Violation of Title 10 CFR 50, Appendix B, Criterion V, having very low safety significance (Green) was self-revealed when a sudden loss of about 23 inches of reactor pressure vessel (RPV) water inventory occurred. An inadequate procedure allowed the operators to create voids which resulted in draining about 2000 gallons of coolant from the RPV into the “B” RHR system when the discharge check valve popped open. This produced a reactor automatic shutdown signal and Group 1 containment isolation on low reactor water level (Level 3) and created the possibility of a water-hammer in the RHR "B" system while the operators were attempting to place RHR "B" in the SDC mode.

Description: As discussed in Section 02.2.b.3, the licensee determined that the top of the heat exchanger was even with the existing RPV water level which resulted in a very small differential pressure across the "B" RHR discharge check valve, 1E12-F031B. This resulted in the check valve only being cracked off its seat - acting more like a flow restriction.

The licensee determined that a combination of a partially opened discharge check valve and an inadequate sequencing of valve manipulations caused the loss of inventory event. Specifically, section 8.1.2.12 of procedure CPS 3312.03, "Warmup of SDC Piping," directed operators to throttle open the RHR "B" to radwaste second isolation valve, 1E12-F040. This valve is downstream of the RHR "B" heat exchanger outlet valve, 1E12-F003B. Opening of the 1E12-F040 valve resulted in draining of the volume between the valve and the 1E12-F003B valve. The operators were then directed to throttle open the 1E12-F003B valve. Because of the flow restriction (partially opened check valve), opening the 1E12-F003B valve resulted in draining piping downstream of the heat exchanger up to the low pressure coolant injection header isolation valve, containment spray header isolation valve, and feedwater header isolation valve. This resulted in steam voids upstream of the heat exchanger and voids in piping downstream of the heat exchanger. When operators opened heat exchanger outlet and bypass valves, void volumes shifted. The licensee believed that the high temperature water and low system pressure and cooling provided by the heat exchanger produced a partial vacuum of about -5 psig at the heat exchanger inlet. The differential pressure between the reactor (20 psig) and the partial vacuum in the heat exchanger caused the check valve to unseat. About 2000 gallons of RPV water drained from the vessel into the SDC piping causing RPV level to drop below the Level 3 setpoint.

In addition, the inspectors concluded that the low pressure condition also contributed to the event. The licensee incorporated the General Electric Service Information Letter (GE SIL) precautions concerning maintaining RPV pressure constant during warmup of the SDC system through a caution statement in CPS 3312.03, "Shutdown Cooling and Fuel Pool Cooling and Assist." Specifically, the caution statement before Step 8.1.2.12 stated "RPV pressure should be held as constant as possible while warming/placing RHR in the SDC mode, to avoid formation of steam voids and water hammer." Because this was a "should" statement and not a "shall" statement, the statement could have been viewed as optional. The inspectors noted that the second operations crew did not maintain pressure; therefore, the inspectors concluded that the licensee did not adequately incorporate the precaution from the GE SIL. The lowering of reactor pressure by the second crew resulted in near saturation conditions. The inspectors concluded that the procedure allowed the operators to create the conditions for void formation and possible water hammer to occur and was therefore deficient.

The inspectors determined that the licensee's root cause team satisfactorily identified the causes leading to the loss of RPV inventory. At the end of the inspection period, the licensee's root cause investigation was not yet complete and therefore corrective actions were not yet finalized. The licensee changed the procedure to allow operators to warmup the SDC system within 100°F of RPV temperature or 200°F heat exchanger temperature. Additionally, the licensee changed the order of valve operations in the warmup section of the procedure to prevent draining sections of piping. The inspectors determined that these conditions will minimize the potential for void formation and water hammer in the SDC system; however, the procedure would not verify adequate differential pressure existed to ensure that the check valve would become unseated during warmup evolutions. During the exit meeting, the licensee stated that another procedure change in progress would ensure that sufficient differential pressure would exist to ensure that the check valve would open.

Analysis: The inspectors determined that the failure to establish adequate procedures for placing the "B" RHR in the SDC mode was a performance deficiency warranting a significance evaluation. Specifically, Section 8.1.2.12 of procedure CPS 3312.03, inadvertently directed the operators to drain portions of the "B" RHR piping downstream of the heat exchanger during the warmup phase. Similarly, the caution statement before Step 8.1.2.12 of procedure CPS 3312.03 did not prevent operators from lowering reactor pressure. As a result, the procedure allowed operators to create voids in the SDC system and was inadequate to prevent a condition that could lead to a water hammer.

The inspectors concluded that this finding had more than minor risk significance in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening, because this condition affected the Reactor Safety Cornerstone, the availability/reliability of mitigating equipment, specifically, the "B" loop of SDC was inoperable after the system was drained and prior to the start of the RHR pump. The inspectors used IMC 0609, "Significance Determination Process," Appendix A, "SDP Phase 1 Screening Worksheet for IE, MS and B Cornerstones," to determine the safety significance of this event. The finding was determined to be of very low safety significance because there was no design deficiency, no actual loss of safety function, no single train loss of safety function for greater than the Technical Specification allowed outage time and no risk due to external events.

Enforcement: Title 10 CFR 50, Appendix B, Criterion V, requires activities affecting quality be prescribed by documented instructions, procedures, or drawing of a type appropriate to the circumstances. The licensee established CPS 3312.03, "Shutdown Cooling and Fuel Pool Cooling and Assist," Revision 4, as the implementing procedure for shutdown cooling. Contrary to the above, on July 14, 2004, CPS 3312.03 was inadequate to prevent the operators from creating conditions that led to the "B" residual heat removal system becoming inoperable. Specifically:

(a) Step 8.1.2.12 directed operators to throttle open the 1E12-F040 (RHR B to radwaste second isolation valve) prior to throttling open the 1E12-F003B (RHR B heat exchanger outlet valve). This resulted in draining water from the discharge side of the heat exchanger resulting in system voiding.

(b) The caution statement before Step 8.1.2.12 stated pressure should be held constant; however, did not require operators to maintain pressure constant or establish a pressure band. This was insufficient to assure that steam voiding in the system would not occur.

Because this violation was of very low risk significance and was captured in the licensee's corrective action program (CR 235832), this violation is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000461/2004007-02). The licensee took immediate corrective action to revise the procedure to hold the reactor pressure constant at a sufficient value to prevent voiding while establishing shutdown cooling, to revise the warmup steps, and to revise the flow path for warmup.

.2 Assessment of Operators' Performance

No findings of significance were identified.

The inspectors determined that, in general, the operators performance was acceptable. The initial operations crews' pre-evolution brief covered the precautions and limitations section of CPS 3312.03, "Shutdown Cooling and Fuel Pool Cooling and Assist." The resultant RPV pressure reduction and flushing operations were conducted in accordance with procedure and management expectations.

However, the inspectors concluded that the performance of the second (relief) crew was less successful. Command and control for the evolution was not well-established. Specifically, the off-going crew held RPV pressure relatively constant as required by a caution statement before Step 8.1.2.12 of the SDC procedure. The caution step warned operators regarding the formation of steam voids in the RHR piping and possible water hammer while decreasing RPV pressure. The need to hold pressure constant was not discussed during the oncoming "A" reactor operator's and control room supervisor's turnover. Unaware of the caution statement, the oncoming crew lowered RPV pressure from 60 psig to about 20 psig to minimize the inoperability time for the suppression pool suction valve (1E12-F004B). [The valve had been declared inoperable due to limitation 6.8 of the SDC procedure and would remain inoperable until reactor coolant temperature was less than 150°F.] The operators responsible for the warmup phase of the procedure were not aware that the reactor operator commenced decreasing reactor pressure. The operators' failure to adhere to the caution statement and the quality of the turnover did not meet site management expectations. The inspectors concluded that the lowering of reactor pressure was a contributor rather than the cause of the event.

The inspectors concluded that re-entering the fill and vent portion of the procedure when a pressure drop was noted and re-entering the warmup portion of the procedure when temperature decreased were acceptable by procedure.

.3 Equipment Concerns

No findings of significance were identified.

With respect to the performance of the "B" RHR pump discharge check valve, 1E12-F031B, the licensee determined that the valve did not open enough to allow sufficient RPV flow into the SDC piping during the warmup phase. The licensee identified that the top of the heat exchanger was even with the existing RPV water level. This resulted in a very small differential pressure to cause the F031B check valve to open. Hence, the valve was only cracked off its seat - acting more like a flow restriction. The inspectors noted that the licensee appropriately opened, inspected and took measurements of the check valve pin and bushings. The measurements taken were within thousandths of an inch of measurements taken 9 years previously. This indicated that the valve was not excessively worn.

As stated in the sequence of events, a second attempt to place "B" RHR in the SDC mode was aborted soon after an in-plant operator identified a puddle of water on the floor of the "B" heat exchanger room near heat exchanger relief valve, 1SX208B. The inspectors concurred with the licensee's belief that the 1SX208B relief valve did not lift. Lifting of this 4-inch valve would have resulted in more water on the floor. The licensee believed thermal changes inside the isolated heat exchanger may have caused lake water system pressure to increase thereby resulting in the valve leaking past its seat. The inspectors determined that operators and radiological protection personnel reacted appropriately to the puddle of water.

Operators in the plant during warmup evolutions and subsequent engineering walkdowns of the system did not identify any evidence that a water hammer had occurred. An evaluation by engineering determined that the resultant pressure spike would not have exceeded the limiting flange strength and that the piping would have remained intact had a pump been started to fill the voided system. The inspectors had no further concerns.

03 Event Common Cause Review and Assessment (93812)

a. Inspection Scope

The team interviewed individuals involved in both events, and reviewed pertinent logs, information, and procedures to identify any common causes or relationships between the two events.

b. Findings

No findings of significance were identified.

The team did not identify any common causes or relationship between the two events.

04 Cross-Cutting Aspects of Findings

A finding described in Section 02.1.b.3 of this report affected the cross cutting area of problem identification and resolution, in that, the effects of the February 2004 feedwater pump modifications on the operators performance of their duties were not identified when preparing the modification package or during the licensee investigation and simulator runs after the March 2004 automatic shutdown.

05 Exit Meeting Summary

On July 28, 2004, the team presented the preliminary inspection results to Mr. R. Bement and other members of the Clinton Power Station management and staff. The licensee acknowledged the information presented. The team asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
R. Bement, Site Vice President
M. McDowell, Plant Manager
J. Cunningham, Work Management Director
R. Davis, Radiation Protection Director
R. Frantz, Regulatory Assurance Representative
M. Hiter, Access Control Supervisor
W. Iliff, Regulatory Assurance Director
J. Madden, Nuclear Oversight Manager
R. Schmidt, Maintenance Manager
D. Schavey, Operations Director
J. Sears, Chemistry Manager
T. Shortell, Training Manager
C. Williamson, Security Manager
J. Williams, Site Engineering Director
R. Zacholski, Shift Operations Superintendent

NRC
A. Stone, Branch Chief, Division of Reactor Projects

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000461/2004007-01	NCV	Failure to Make Plant Personnel Aware of a Modification Which May Affect the Performance of Their Duties
05000461/2004007-02	NCV	Failure to Have an Adequate Operating Procedure

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any pat of it, unless this is stated in the body of the inspection report.

CPS 1041.01F002; Post Trip Review Report Routing
CPS 3312.03, Shutdown Cooling and Fuel Pool Cooling and Assist, Revision 4
CPS 3312.03, Shutdown Cooling and Fuel Pool Cooling and Assist, Revision 4a
CPS 4002.01; Abnormal RPV Level/Loss of Feedwater At Power
CPS 4100.01; Reactor An automatic
CPS 9000.06, Shutdown Cooling Temperature Data Sheet, Revision 31a
CR 236745; Unexpected Inleakage During Draining Evolutions of RHR “B”
CR 236746; Unexpected Inleakage During Draining Evolutions of RHR “B”
CR 210048; Turbine Driven Reactor Feed Pump “B” Tripped on the SCRAM Transient.
CR 210808; Feedwater System Transient Response To SCRAM on 3/22/04
CR 237467; Clean Contacts In RR “B” 2 Breaker Circuit
CR 237898; Potential Operator Workaround - TDRF pump Trip Logic
CR 237792; FW Contingency Plan For Loss/Potential Loss Of MDRP-RX SCRAM
CR 235823, Reactor Vessel Drain Down Event - Investigation Report, July 14, 2004
CR 235823, Reactor Vessel Drain Down Event - Troubleshooting Log, July 14, 2004
MA-AA-716-004; Troubleshooting Log
EC 338996, Revision 000; Modify Timing of TDFW Pump Trip On Low Suction Pressure So That Only One Feed Pump Is Tripped At A Time - To Avoid SCRAM
Clinton Power Station 7-13-04 Plant Trip Event Summary
Restart Plant Oversight Review Committee Presentation 7-16-04
Prompt Investigation Report, Clinton Power Station
Exelon HU-AA-104-101, Procedure Use and Adherence, Revision 0
Daily Station Dose (TE007), July 13 - 14, 2004
WO 717869, Task 1, Open/Inspect RHR B Pump Discharge Check Valve, July 17, 2004
Clinton Power Station Control Room Logs, July 13 & 14, 2004
GE SIL 175, RHR/Recirculation System Water Hammer During Primary System Cooldown, June 15, 1976

<u>Drawings/Charts Reviewed</u>
MO5-1075, P&ID Residual Heat Removal, Revision AV
Data Acquisition Recorder trend charts, July 13 & 14, 2004

LIST OF ACRONYMS USED

ADAMS	Agency wide Documents Access and Management System
CPS	Clinton Power Station
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
ESW	Emergency Service Water
FWLCS	Feedwater Leakage Control System
FPC&A	Fuel Pool Cooling and Assist
GCB	Gas Circuit Breakers
GE SIL	General Electric - Service Information Letter
IMC	Inspection Manual Chapter
LPCI	Low Pressure Coolant Injection
MCR	Main Control Room
MDRF	Motor Driven Reactor Feedwater
MPT	Main Power Transformer
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PMT	Post Maintenance Testing
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RR	Reactor Recirculation
SDC	Shut Down Cooling
SDP	Significance Determination Process
TDRF	Turbine Driven Reactor Feedwater
USAR	Updated Safety Analysis Report

July 15, 2004

MEMORANDUM TO: Carey Brown, Resident Inspector, Clinton Power Station

FROM: Ann Marie Stone, Chief, Reactor Projects Branch 3 ***/RA/***

SUBJECT: SPECIAL INSPECTION CHARTER FOR CLINTON POWER STATION, UNIT 1 REACTOR SCRAM AND LEVEL TRANSIENT ON JULY 13 AND 14, 2004

At 1608 on Tuesday, July 13, 2004, an automatic scram occurred from 95 percent power (which is the maximum power achievable under the authorized power uprate) due to an apparent instantaneous neutral phase over current fault trip on the main power transformer. All plant systems operated normally on the scram with the exception of the "A" recirculation pump that ran back per design and failed to restart on low speed (15 cycles.) In addition, during the recovery, the "B" feedwater pump tripped on low suction pressure when operators secured the "A" feedwater pump.

Further complications occurred on Wednesday, July 14, 2004, at 0045 when the reactor pressure vessel level dropped about 24 inches resulting in another reactor protection system and isolation actuation. Prior to the second event, the unit was in Mode 3 with reactor pressure at 18 psig and level stable at 32 inches. The licensee was in the process of preparing the "B" residual heat removal (RHR) system for the shutdown cooling mode of operation. The licensee was heating the heat exchanger portion of the system through natural circulation using a flow path from the reactor vessel to the radwaste system. Once secured, the licensee verified the system was filled and vented; however, pressure and temperature in the heat exchanger decreased. When the licensee re-established the heating sequence, reactor vessel level dropped. The licensee believes that the "B" RHR discharge check (1E12-F031B) was stuck off its seat which caused water to drain from the system. The licensee made two additional attempts to prepare the "B" RHR system for shutdown cooling; however, when similar indications of a loss of temperature and pressure in the discharge piping occurred, further attempts to prepare the system were stopped.

Using the deterministic criteria provided in Management Directive 8.3 and Inspection Procedure 71153 "Event Followup," a special inspection will be conducted. Specifically, the unexpected system interactions that resulted in the loss of coolant inventory, and the multiple electrical equipment failures, could indicate concerns with licensee operational performance. The special inspection will be performed by you, as Team Leader; Keith Walton, Operations Engineer, and Daneira Melendez-Colon, Nuclear Safety Professional. The special inspection will evaluate the facts, circumstances, and licensee actions surrounding the events. A charter is attached. The nominal duration of the inspection is expected to be approximately 4-6 days.

Attachment: As stated

cc w/att: J. Caldwell, ORA
G. Grant, ORA
S. Reynolds, DRP
P. Hiland, DRP
R. Caniano, DRS
C. Pederson, DRS
D. Pickett, NRR
R. Lanksbury, DRS
B. Dickson, Clinton
K. Walton, DRS
D. Melendez-Colon, Clinton

SPECIAL INSPECTION (SI) TEAM CHARTER

Clinton Power Station Reactor Scram and Complications on July 13 and 14, 2004

The Special Inspection should focus on the following:

1. Adequacy of the licensee's evaluation of the July 13, 2004 reactor scram which includes:

    - the cause of the turbine-generator trip which resulted in the reactor trip;
    - unexpected response of the gas circuit breaker 4510;
    - evaluation of the effects of the scram including associated potential damage to the main power transformer and switchyard equipment;
    - the cause of the "A" recirculation pump failing to shift to low speed as designed; and
    - the cause of the "B" feedwater pump trip including an evaluation of effectiveness of a modification installed in February 2004 to prevent such trips.

2. Adequacy of the licensee's evaluation of the July 14, 2004 event which includes:

    - an assessment of operators' performance during the preparation to place "B" RHR system in shutdown cooling mode of operation;
    - an assessment of the procedures used to place "B" RHR system in the shutdown cooling mode of operation including the decision to proceed following the initial temperature and pressure anomalies;
    - the cause for the loss of water inventory and subsequent inability to place the system in shutdown cooling; and
    - the unexpected performance of the "B" RHR discharge check valve, the RHR heat exchanger service water relief valve, and other equipment concerns identified during the course of the licensee's investigation.

Charter Approval

/RA/
Chief, Reactor Projects Branch 3

/RA/
Director, Division of Reactor Projects