December 1, 2000

Mr. Michael T. Coyle
Vice President
Clinton Power Station
AmerGen Energy Company, LLC
Mail Code V-275
P. O. Box 678
Clinton, IL 61727

SUBJECT: CLINTON POWER STATION - NRC INSPECTION REPORT 50-461/00-17(DRP)

Dear Mr. Coyle:

On November 13, 2000, the NRC completed an inspection at your Clinton Power Station. The enclosed report documents the inspection findings which were discussed on November 13, 2000, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and 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.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green). Both of these issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Clinton facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS)*. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas J. Kozak, Chief
Reactor Projects Branch 4

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

Enclosures: Inspection Report No. 50-461/00-17(DRP)

cc w/encl: P. Hinnenkamp, Plant Manager
M. Reandeau, Director - Licensing
G. Rainey, Chief Nuclear Officer
E. Wrigley, Manager-Quality Assurance
M. Aguilar, Assistant Attorney General
G. Stramback, Regulatory Licensing
Services Project Manager
General Electric Company
Chairman, DeWitt County Board
State Liaison Officer
Chairman, Illinois Commerce Commission

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REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	50-461/00-17(DRP)
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Route 54 West Clinton, IL 61727
Dates:	October 1 - November 13, 2000
Inspectors:	P. L. Louden, Senior Resident Inspector C. E. Brown, Resident Inspector L. L. Collins, Project Engineer D. E. Zemel, Illinois Department of Nuclear Safety
Approved by:	Thomas J. Kozak, Chief Reactor Projects Branch 4 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
•Initiating Events •Mitigating Systems •Barrier Integrity •Emergency Preparedness	•Occupational •Public	•Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

Clinton Power Station
NRC Inspection Report 50-461/00-17(DRP)

IR 05000461-00-17, on 10/01-11/16/2000; AmerGen Energy Company LLC; Clinton Power Station; refueling and outage activities; event followup.

The inspection was conducted by the resident inspectors and a regional projects inspector. Two Green findings were identified during the inspection, both of which were treated as Non-Cited Violations. The significance of all 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 are indicated by “no color” or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- GREEN. During replacement of power supplies for the alternate rod insertion (ARI) system, maintenance personnel failed to fully evaluate the impacts that re-energizing the power supplies had on the ARI initiation logic. While re-energizing the power supplies, the initiation logic sensed an ARI signal (low reactor water level). This caused the vent and drain valves to close and the scram discharge volume to fill with water. Plant operators inserted a manual scram signal before the automatic high scram discharge volume set point was reached. One Non-Cited Violation was identified for having an inadequate maintenance procedure to control this activity. (Section 1R20)

  This finding was evaluated using the shutdown significance determination process contained in Appendix G of IMC 0609 and was determined to have very low risk significance because it did not impact any of the five shutdown safety functions identified by NUMARC 91-06.

Cornerstone: Barrier Integrity

- GREEN. Secondary containment was inoperable for 6 minutes during fuel movements when secondary containment interlock doors were inadvertently opened to move scaffolding. The inoperability was discovered when operators in the control room received an alarm indicating a loss of secondary containment vacuum. One Non-Cited Violation was identified for violating Technical Specification 3.6.4.1 which requires secondary containment operability during fuel moves. (Section 4OA3)

  This finding was evaluated using the shutdown significance determination process contained in Appendix G of IMC 0609 and was determined to have very low risk significance because it did not meet the criteria for findings requiring a phase 2 significance evaluation.

Report Details

Summary of Plant Status

At the beginning of the inspection period, the plant was in a power coast down toward the scheduled refueling outage which commenced on October 14. Following completion of the refueling outage, the plant was restarted on November 10 and connected to the electrical distribution grid on November 12.

1. Reactor Safety

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors reviewed piping and instrumentation drawings (P&IDs) and conducted partial walkdowns of the safety significant systems listed below. The walkdowns were conducted to verify equipment alignment and to identify any discrepancies that could impact the system function.

- Emergency Diesel Generators (EDGs) and Support Systems, referencing CPS 3506.01E001, “Diesel Generator and Support Systems Electrical Lineup,” Revision 17 and CPS 3506.01V001, “Diesel Generator and Support Systems Valve Lineup,” Revision 10

- Control Room Heating Ventilation and Air Conditioning (HVAC), referencing CPS 3402.01V001, “Control Room HVAC Valve Lineup,” Revision 15a

- Station Service Air System, referencing CPS 3214.01E001 & CPS 3214.01V001, “Plant Air Electrical and Valve Lineups”, Revisions 9 and 20 respectively

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed portions of the licensee’s Fire Protection Evaluation Report (FPER) and the Updated Safety Analysis Report (USAR) to verify consistency in the documented analysis with installed fire protection equipment at the station. To assess the control of transient combustibles and ignition sources, the material and operational condition of fire-protection systems and equipment, and the status of fire barriers, the inspectors conducted walk downs of the following risk significant areas:

- Primary Containment

- Emergency Core Cooling System (ECCS) Rooms
- Auxiliary Building Steam Tunnel
- Reactor Drywell
- Safety Related Battery Rooms
- Turbine Building including heater bay

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee’s implementation of the maintenance rule (MR) requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, and current equipment performance problems. The following systems were selected based on their designation as risk significant under the MR, or because of increased monitoring required by MR category a(1).

- Reactor Core Isolation Cooling System with a focus on the assessment of system steam supply admission valve repetitive MR functional failures
- Instrument Air System with a focus on the assessment of a containment isolation air operated valve/solenoid operated valve functional failure

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors observed the licensee’s risk assessment processes and considerations used to plan and schedule maintenance activities on safety-related structures, systems, and components particularly to ensure that maintenance risk and emergent work contingencies had been identified and implemented. The inspectors assessed the effectiveness of risk management activities for the following work activities or work weeks:

- Risk planning and administrative controls when the shutdown cooling mode of the residual heat removal (RHR) system was out-of-service and the fuel pool cooling system was used for decay heat removal.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability determinations and evaluations affecting mitigating systems to ensure that operability was properly justified and the component or system remained available such that no unrecognized risk increase had occurred.

- CR 2-00-09-113 - Engineering Evaluation and calculation IPQ451 associated with a Division III EDG seismic qualification concern

- CR 2-00-11-034 - Division II battery charger regulation band out of tolerance

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed and observed portions of the following post maintenance testing (PMT) activities involving risk significant equipment to ensure that the activities were adequate to verify system operability and functional capability.

- Testing evaluations associated with “ping” testing of primary reactor coolant piping to verify functionality of loose parts monitoring detectors.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors conducted inspections of the following areas during the October/November 2000 refueling outage:

- Outage risk assessment of planned activities to verify defense-in-depth would be maintained and high risk activities were appropriately controlled and reviewed by station management.

- Operational evolutions such as, plant shutdown, establishing shutdown cooling, reactor startup, and turbine-generator electrical distribution grid synchronization.

- Component and equipment configuration management control to ensure equipment relied on to perform a key safety function would not be adversely affected by outage activities.

- Clearance and special operating permit programs.

- Reactor coolant system instrumentation to ensure operators maintained a clear understanding of accuracy of measurement and contingencies if the instrument indications were lost.

- Decay heat removal system operability and protection during key times of the outage, and during special surveillance testing.

- Containment integrity control as required.

- Review of selected outage related maintenance and surveillance activities to ensure the activities were conducted in accordance with station procedures and Technical Specification (TS) requirements.

- Reactor restart activities including approach to critical, turbine startup, recirculation system motor speed change, and ascension to 100 percent power.

b. Findings

During the refueling outage, the licensee replaced power supplies for the ARI system as part of a planned overhaul of many in-plant power supplies. The preventive maintenance work tasks (PCIMSA017 and PCIMSA019) were first time evolutions. On October 16, 2000, maintenance personnel re-energized power supplies for the alternate rod insertion (ARI) system. During this task, an ARI initiation signal was received by the system logic which caused the scram discharge volume vent and drain valves to close. With the valves closed, the scram discharge volume began filling up. Control room operators inserted a manual scram signal to the reactor protection system (RPS) before the automatic scram set point for high water level in the scram discharge volume was reached. The event was reported pursuant to 10 CFR 50.72 as an unplanned valid RPS actuation. Licensee Event Report (LER) 50-461/2000-006 was submitted documenting the event and planned corrective actions.

It was identified during the licensee’s review of the event that the impact of re-energizing the ARI system power supplies was not adequately evaluated and that deficiencies existed in the power supply installation work plan and procedures. The inspectors considered this procedure deficiency to be more than minor because, if left uncorrected, other unplanned RPS actuations could occur which could increase the likelihood of a plant transient or scram. This finding was evaluated using the shutdown significance determination process contained in Appendix G of IMC 0609. The finding was classified as having very low risk significance (Green) because it did not impact any of the five shutdown safety functions identified by NUMARC 91-06.

Technical Specification 5.4.1.a requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, dated February 1978. Appendix A to RG 1.33 recommends procedures for performing maintenance be implemented for maintenance that could affect the performance of safety-related equipment. Preventive maintenance tasks PCIMSA017 and PCIMSA019, "Replacement of ATWS [anticipated transient without scram] system-1 "A" and "B" power supplies," are procedures for performing maintenance that could affect the performance of safety-related equipment. Contrary to this requirement, PCIMSA017 and PCIMSA019 were inadequate for the maintenance activity performed because the effects that the power supply restoration had on safety-related equipment were not adequately planned and evaluated. However, because of the very low safety significance of this issue and because the licensee has included this issue in their corrective action program (CR 2-00-10-083) this procedure violation is being treated as a Non-Cited Violation **(NCV 50-461/00-17-01)**.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed portions of the following surveillance tests to verify that risk significant systems and equipment were capable of performing their intended safety functions and assessed their operational readiness:

- CPS 9080.23, "Diesel Generator 1C - ECCS Integrated, " Revision 26b
- CPS 9382.17, Division II 125Vdc Battery Capacity Test," Revision 25a
- CPS 9080.22, "Diesel Generator 1B - ECCS Integrated," Revision 25a
- Residual Heat Removal and Main Steam System valve operability testing

b. Findings

No findings of significance were identified.

4. Other Activities

4OA1 Performance Indicator Verification

Reactor Coolant System Leakage (71151)

a. Inspection Scope

The inspectors reviewed the data collection and methods used to calculated identified and unidentified reactor coolant system leakage. The inspectors verified that the total of both leak rates (in gallons/minute) was summed and expressed as a percentage of the TS allowable total leak rate of 30 gallons/minute.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) LER 50-461/2000-005: “Losses of Secondary Containment Integrity During Movement of Irradiated Fuel Due to Unauthorized Opening of Boundary Doors and Failure to Adequately Verify Local Leak Rate Test Boundary Valve Position”. The inspectors reviewed the circumstances surrounding the LER and determined that the events were of very low risk significance (GREEN). Secondary containment was inoperable during fuel movements which violated Technical Specification 3.6.4.1. On one occasion, secondary containment interlock doors were inadvertently opened causing a loss of secondary containment vacuum and an alarm in the control room. Operators responded by halting fuel moves and restoring secondary containment operability within 6 minutes. Using the phase 1 shutdown significance determination process, this finding was determined to have very low risk significance because the criteria for findings requiring a phase 2 analysis were not met. Three additional occurrences of the loss of secondary containment integrity occurred during the refueling outage and were reported in the LER, but were more minor in nature. In all cases, once identified, operators took the appropriate actions to secure from fuel movements or core alterations. The events have been placed in the licensee’s corrective action program as CRs 2-00-10-172, 2-00-10-217, 2-00-10-230, and 2-00-10-238. However, because of the very low safety significance of this issue and because the licensee has included this issue in their corrective action program this violation is being treated as a Non-Cited Violation **(NCV 50-461/00-17-02)**.

.2 (Closed) LER 50-461/2000-006: “Lack of System Response Characteristics Knowledge Results in Failure to Identify Consequences of Re-Energizing Power Supplies and Manual Reactor Scram”. This event was discussed in Section 1R20 of this report. No additional findings of significance were identified.

4OA6 Meetings, including Exit

The inspectors presented the inspection results to Mr. M. T. Coyle, Site Vice President, and other members of licensee management on November 13. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

S. Clary, Director - Plant Engineering
M. Coyle, Site Vice President
W. Iliff, Director - Experience Assessment and Corrective Actions
P. Hinnenkamp, Plant Manager - Clinton Power Station
W. Maguire, Director - Operations
R. Moore, Manager - Work Management
A. Plater, Radiation Protection Manager
M. Reandeau, Director - Licensing
R. Schenck, Manager - Maintenance
D. Smith, Director - Security and Emergency Planning
P. Walsh, Manager - Nuclear Station Engineering Department
E. Wrigley, Manager - Quality Assurance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-461/00-17-01	NCV	Inadequate procedures associated with the re-energizing of ARI power supplies which led to an unplanned ARI initiation while the reactor was shutdown.
50-461/00-17-02	NCV	Licensee identified violation of T.S. 3.6.4.1 which requires secondary containment to be operable during fuel movement and core alterations.

Closed

50-461/00-17-01	NCV	Inadequate procedures associated with the re-energizing of ARI power supplies which led to an unplanned ARI initiation while the reactor was shutdown.
50-461/00-17-02	NCV	Licensee identified violation of T.S. 3.6.4.1 which requires secondary containment to be operable during fuel movement and core alterations.
LER 50-461/2000-005		Licensee identified violation of T.S. 3.6.4.1 which requires secondary containment to be operable during fuel movement and core alterations. (NCV 50-461/00-17-02)
LER 50-461/2000-006		Inadequate procedures associated with the re-energizing of ARI power supplies which led to an unplanned ARI initiation while the reactor was shutdown. (NCV 50-461/00-17-01)

Discussed

None

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ARI	Alternate Rod Insertion
CPS	Clinton Power Station
CR	Condition Report
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FPER	Fire Protection Evaluation Report
HVAC	Heating, Ventilation and Air Conditioning
LER	Licensee Event Report
IMC	Inspection Manual Chapter
MR	Maintenance Rule
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
P&ID	Piping and Instrumentation Drawing
PARS	Publicly Available Records
PERR	Public Electronic Reading Room
PMT	Post Maintenance Testing
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
SDP	Significance Determination Process
TS	Technical Specification
USAR	Updated Safety Analysis Report