July 17, 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 - NRC INSPECTION REPORT 50-461/2000012(DRS)

Dear Mr. Coyle:

On June 16, 2000, the NRC completed the baseline annual inspection of Evaluations of Changes, Tests, or Experiments (10 CFR 50.59) and the baseline biennial Permanent Plant Modifications inspection at your Clinton Power Facility. The enclosed report presents the results of that inspection which were discussed on June 16, 2000, with you and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to changes to facility structures, systems, and components, normal and emergency procedures, and the Updated Safety Analysis Report in accordance with the requirements of 10 CFR 50.59; and changes to the facility via permanent plant modifications to verify compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of design documents, procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, two issues of very low safety significance (Green) were identified. The two issues were examples of a violation of NRC regulations which involved inadequate post-modification testing requirements and failure to perform required post-modification testing on the emergency diesel generators. However, the violations were not cited due to their very low safety significance and because they have been entered into your corrective action program. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with a copy to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 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 by Steve Reynolds Acting For/

John A. Grobe, Director
Division of Reactor Safety

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

Enclosure: Inspection Report 50-461/2000012(DRS)

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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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	50-461/2000012(DRS)
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Route 54 West Clinton, IL 61727
Dates:	June 12-16, 2000
Inspectors:	Z. Falevits, Reactor Engineer, Lead Inspector J. Gavula, Reactor Engineer B. Scott, Reactor Engineer
Approved by:	Ronald N. Gardner, Chief, Electrical Engineering Branch Division of Reactor Safety

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

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- 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 safety 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 50-461/2000012(DRS)

The inspection was a baseline annual inspection of Evaluations of Changes, Tests, or Experiments (10 CFR 50.59) and a baseline biennial Permanent Plant Modifications inspection. The inspection was conducted by three regional engineering specialists. The inspection was an examination of activities conducted under your license as they relate to changes to facility structures, systems, and components, normal and emergency procedures, and the Updated Safety Analysis Report in accordance with the requirements of 10 CFR 50.59; and changes to the facility via permanent plant modifications to verify compliance with the Commission's rules and regulations and with the conditions of your license. The inspection identified two green issues which were considered examples of a non-cited violation. The significance of issues is indicated by their color (green, white, yellow and red) and was determined by the Significance Determination Process.

Cornerstone: Mitigating Systems

Green: The licensee failed to ensure that the appropriate post-modification testing (PMT) was specified in the Division I and Division III emergency diesel generator (EDG) output breaker circuitry modification packages and that the post-modification tests were correctly accomplished. This was required to demonstrate through component and functional testing that the modified (rewired) portions of the Division I and Division III EDG output breaker circuitry were adequately installed to accomplish the intent of the plant design changes. This is considered two examples of a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as CR 2-00-06-084 and CR 2-00-06-090.

These deficiencies had very low safety significance (green). In both cases, there was an extremely low probability of a simultaneous occurrence of a Loss of Coolant Accident, Loss of Offsite Power, and failure of either EDG output breaker to close. In the case of the Division III EDG output breaker, while the licensee had not conducted an auto-start test of the EDG, other testing, such as circuit voltage measurements and ground testing had been conducted. Further, the licensee stated that the portions of the circuit containing the Loss of Offsite Power contacts had not been disturbed during the modification. Finally, portions of the circuit were tested during a manual breaker closure surveillance. All of these provided reasonable assurance that the Division III EDG output breaker would function if called upon. In the case of the Division I EDG output breaker, the other train of mitigating systems, powered by the Division II EDG was operable and operators could take manual action to close the Division I EDG output breaker, restoring power to the Division I systems.

Report Details

1. REACTOR SAFETY

Cornerstone: Mitigating Systems and Barrier Integrity

1R02 Evaluations of Changes, Tests and Experiments (IP 71111, Attachment 2)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

The inspectors reviewed 10 evaluations performed pursuant to 10 CFR 50.59, one of which pertained to the barrier integrity cornerstone. The evaluations related to permanent plant modifications, setpoint changes, procedure changes, conditions adverse to quality, and changes to the Updated Final Safety Analysis Report. The inspectors also reviewed 14 evaluation applicability checks where the licensee had determined that a 50.59 evaluation was not necessary. In regard to the changes reviewed where no 50.59 evaluation was performed, the inspectors verified that the changes were minor editorial clarifications that did not meet the threshold of a “change to the facility as described in the safety analysis report.” For the 50.59 evaluations, the inspectors confirmed that prior NRC approval was obtained or was not required for the implemented changes.

b. Findings

There were no findings identified.

1R17 Permanent Plant Modifications (IP 71111, Attachment 17)

.1 Review of Recent Plant Modifications

a. Inspection Scope

The inspectors reviewed 12 plant modifications that were installed in the last several years. The packages were chosen based upon their affecting systems that had high Maintenance Rule safety significance or high risk significance in the licensee's Individual Plant Evaluation. All of the modifications involved changes to mitigating systems, while one affected both mitigating systems and barrier integrity. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, it's associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

b. Findings

Inadequate Post-Modification Testing of Electrical Modifications

(1) Modification to Division III High Pressure Core Spray EDG Output Breaker Circuitry

Condition Report (CR) 1-98-09-017, dated September 1, 1998, documented that the design of the control circuit for the Division III EDG output breaker included over 3000 feet of #12 AWG wire. Consequently, excessive voltage drop in the auto start circuitry wiring resulted in the closing coil for the Division III EDG breaker receiving less than the vendor - required voltage of 90 Vdc. Failure to provide the rated voltage to the closing coil could have resulted in the Division III EDG output breaker failing to close with a resultant loss of power to the 4.16KV bus during a loss of offsite power (LOOP). This problem has existed since original construction.

To address this design deficiency, Engineering Change Notice (ECN) 31064 was issued on November 6, 1998. The ECN re-wired various portions of the control circuitry in order to reduce the length of the cable and the overall resistance in the circuit. This was accomplished by shortening the circuit, paralleling, and re-terminating conductors.

The inspectors reviewed the ECN, related design drawings, and the specified and accomplished post-modification testing to determine if the modification was correctly designed, installed and tested before the Division III EDG was declared operable in January 1999. The inspectors determined that Part II of Clinton Power Station No. 1003.01F002, “Detailed Impact Assessment” in ECN 31064 contained conflicting and inconsistent requirements for post-modification testing of this modification, as follows:

- Plant operations specified performance of surveillance 9080.02, “Diesel Generator 1C Operability - Manual and Quick Start Operability” for manual closing PMT and portions of 9080.23, “Diesel Generator lC - ECCS Integrated” for auto closing of PMT.

- Testing and Installation engineer stated, “The complexity of the changes made by this PC ECN may require a special test to verify the existing circuit logic has been maintained. CPS No. 9080.02 and 9080.23 need to be performed as part of the PMT to verify logic is functional.”

- System engineers stated that testing requirements are adequate to verify wiring changes.

In addition, CPS No. 1003.01F002, Part III, “Installation and Testing Requirements” Section 2.3 specified testing requirements to verify design as (1) verify the manual closing capability of the breaker from Main Control Room (MCR) and DG control panel, and (2) verify (a) the automatic closing function of the breaker and (b) the blocking of the function when permissives are not established. Section 2.5 specified the functional testing acceptance requirements as (1) MCR and local panel hand switches can close breaker, and (2) breaker closes automatically in response to LOOP scenario but does not close in if all permissives are not established.

Maintenance Work Request (MWR) D80263, dated January 15, 1999, specified under “Operations Testing Requirements,” the following post-modification tests: (1) verify installation is correct, and (2) perform satisfactory closing of the Division III EDG output breaker IAW 9080.02 or 9080.23. The MWR documented that the breaker was satisfactorily closed during performance of 9080.02 (manual breaker close) on January 13, 1999.

The inspectors determined that no testing was performed to verify the automatic closing function of the Division III breaker for a LOOP nor was testing performed to verify wiring integrity of the control circuits affecting the manual closing function from the EDG local control panel. Consequently, the ability of the Division III EDG output breaker to automatically close on a LOOP was indeterminate.

On June 15, 2000, the licensee declared the Division III EDG inoperable but available, issued CR No. 2-00-06-084, and for immediate corrective action verified voltage available at the end point of the rewired breaker auto circuitry. The inspectors were informed that the test indicated that +63.5 Vdc was available at point 4 of relay 74F1-2 of the auto-close circuitry.

On June 22, 2000, the licensee performed an engineering evaluation No. 2-00-06-084 to assess the immediate corrective action taken for CR 2-00-06-084, dated June 15, 2000. The evaluation concluded that the control circuit for Division III EDG output breaker was capable of closing the EDG feed breaker in the event of a LOOP. Rationale provided for this conclusion included: (1) the portion of the auto close logic that contained the LOOP permissive contacts was not disturbed during the modification and was tested in August 1998 via surveillance CPS 9080.23 so it did not have to be tested, (2) portions of the auto close circuit is common to the MCR manual operation and LOOP auto operation of the breaker and was retested during the manual breaker closure test performed via CPS 9080.02 in January 1999, and (3) the verification of DC power at terminal 4 of relay 74F1-2 of the auto closing circuit of the Division III output breaker.

The inspectors informed the licensee that since the required functional test (CPS 9080.23) was not performed after the modification was installed in January 1999, as required by ECN 31064, the portion of the auto close circuitry that was not tested has not been positively proven to be fault free.

The inspectors used the significance determination process to evaluate the significance of the failure to conduct post-modification testing of the Division III emergency diesel generator and concluded that the finding was of very low safety significance (green). This finding had the potential to impact mitigating systems (High Pressure Core spray). However, there was an extremely low probability of a simultaneous occurrence of a Loss of Coolant Accident, Loss of Offsite Power, and failure of the Division III EDG output breaker to close. Also, while the licensee had not conducted an auto-start test of the EDG, other testing, such as circuit voltage measurements and ground testing, had been conducted. Further, the licensee stated that the portions of the circuit containing the Loss of Offsite Power contacts had not been disturbed during the modification. Finally, portions of the circuit were tested during a manual breaker closure surveillance. All of these provided reasonable assurance that the Division III EDG output breaker would function if called upon.

Failure to perform required post-modification testing is an example of a Violation of 10 CFR 50, Appendix B, Criterion XI. This violation is considered a Non-Cited Violation (50-461/20012-01a), consistent with the General Statement of Policy and Procedure for NRC Enforcement Actions (NUREG 1600) (Enforcement Policy), Section VI.A.1. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as CR-2-00-06-084. The licensee committed to perform additional testing of this modification (negative test on K1 and K5 relay contacts) during the upcoming Division III EDG outage in September 2000, to ensure positive function of the lockout interlocks in the breaker control circuitry.

(2) Modification to Division 1 EDG Output Breaker Circuitry

ECN 30253, dated August 1, 1997, added remote shutdown panel (RSP) switch 1C61-HS501 isolation contacts (71-71C and 73-73C) into the Division I EDG 4.16 kV feed breaker (252-DG1KA) control circuitry. This design change was required by an Appendix “R” Safe Shutdown Analysis (SSA). The SSA credits use of the EDG on-site source during safe shutdown from the RSP in the event of a MCR- fire. The purpose of the design change was to ensure that in the event of a fire in the MCR, the EDG breaker control circuitry and backup fuses located in the RSP will be isolated from the MCR circuitry. This electrical isolation is needed to prevent potential damage to the Division I EDG output breaker 125 Vdc control power circuitry and backup control power fuses. These RSP backup fuses and control circuitry are relied upon to provide 125 Vdc control power to the EDG feed breaker following a MCR fire.

The inspectors conducted a thorough review of this plant change (modification) and determined that post-modification testing was not accomplished to demonstrate opening of the switch contacts by switch operation and to verify through component and functional testing that the rewired portion of the EDG circuitry was adequately installed to accomplish the intent of the design change. The inspector determined that the only post-modification testing requirements specified by the design engineer in the ECN package was to “verify circuit integrity after installation” (item 21 of CPS No. 1003.01F005). MWR No. D50576 only verified continuity of the two newly-wired contacts. Post-modification testing was not performed to ascertain functional confirmation of the design change and to verify the isolation function of the switch contacts.

The inspectors reviewed the licensee’s self-assessment report 2000-026, “Review of RF-6 Post-Modification Testing” dated March 20, 2000. This assessment report documented results of an engineering assessment performed in March and April 1999 to evaluate post-modification testing of 187 refueling outage 6 (RF-6) hardware changes. Of the 187 packages evaluated, 182 were found to have performed acceptable PMT. Five packages were deemed unacceptable. The licensee informed the inspectors that ECN 30253 (see above writeup) was one of the 187 packages examined during the licensee's assessment and the PMT for this ECN was determined to be "weak". In addition, the licensee stated that a number of Condition Reports were issued in 1998 and 1999 that questioned the adequacy and implementation of PMT at Clinton.

The inspectors used the significance determination process to evaluate the significance of the failure to specify and conduct proper post-modification testing of the Division I emergency diesel generator and concluded that the finding was of very low safety significance (green). This finding had the potential to impact mitigating systems (Emergency Power). However, there was an extremely low probability of a simultaneous occurrence of a Loss of Coolant Accident, Loss of Offsite Power, and failure of the Division I EDG output breaker to close. Further, the other train of mitigating systems, powered by the Division II EDG was operable. Finally, operators could take manual action to close the Division I EDG output breaker, restoring power to the Division I systems.

Failure to properly specify and perform required post-modification testing is considered an example of a Violation of 10 CFR 50, Appendix B, Criterion XI. This violation is considered a Non-Cited Violation (50-461/20012-01b), consistent with the General Statement of Policy and Procedure for NRC Enforcement Actions (NUREG 1600) (Enforcement Policy), Section VI.A.1. This violation was identified by the NRC and promptly entered by the licensee into the corrective action program as CR-2-00-06-090. The licensee committed to perform additional testing of this modification during the next Division I EDG outage scheduled for the week of August 7, 2000.

4. **OTHER ACTIVITIES**

4OA6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Coyle, Vice President, and other members of licensee management at the exit meeting held on June 16, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Coyle, Vice President
P. Hinnenkamp, Plant Manager - Clinton Power Station
M. Reandeau, Director - Licensing
K. Baker, Director Design Engineering
E. Halverson, Supervisor Mechanical Design
M. Norris, Supervisor Engineering Assurance
P. Walsh, Manager - Engineering Department
E. Wrigley, Manager - Quality Assurance
K. Gallogly, Director Experience Assessment
V. Cwietniewicz, Manager Nuclear Training
J. Forman, Licensing Engineer

NRC

R. Gardner, Chief, Electrical Engineering Branch, Division of Reactor Safety
C. Brown, Resident Inspector

INSPECTION PROCEDURES (IPs) USED

IP 71111.02	Changes to License Conditions and Safety Analysis Report
IP 71111.17	Permanent Plant Modifications
IP 71150	Plant Status (reference)
IP 71152	Identification and Resolution of Problems (reference)

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Closed

None.

Discussed

None.

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CPS	Clinton Power Station
CR	Condition Report
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
ECN	Engineering Change Notice
IP	Inspection Procedure
IR	Inspection Report
LOOP	Loss of Offsite Power
MCR	Main Control Room
MWR	Maintenance Work Request
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PERR	Public Electronic Reading Room
PMT	Post-Modification Testing
RSP	Remote Shutdown Panel
SE	Safety Evaluation
TS	Technical Specification
USAR	Updated Safety Analysis Report

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but, rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort.

Calculations

EMD-052581, Addendum F, Piping Evaluation for Subsystem 1SX19 of Shutdown Service Water System due to the Installation of Vacuum Breaker Spray Deflector 1SX26MB, Revision 1, May 6, 1999.
IP-M-0273, Verification of Water Hammer Assumptions, Revision 1, May 11, 1999.
IP-M-0457, Water Hammer Potential of SX System Division 3, Revision 0, January 21, 1998.
IP-M-308, SX Pressure Transients for CR 1-93-05-011, Revision 0, July 17, 1993.
19-G-3, Cable Ampacities in Ducts, May 13, 1976.
19-AN-14, Division I and II Diesel Generator Relay Settings, January 25, 2000.

Condition Reports

1-98-10-139, Safety evaluation deficiency for ECN 30860, November 9, 1998.
1-99-05-178, Lack of rigor in justification for ECN 31678, May 20, 1999.
1-99-06-001, Valve 1CC321A (installed by ECN 31029) is SStl vs carbon steel, May 29, 1999.
2-00-06-081, ECN 30532 did not identify all affected documents, June 15, 2000.
2-00-06-084, Inadequate Implementation of Post-Modification Testing Requirements, June 15, 2000.
1-98-09-017, Control Circuit for Division III DG Feed Breaker to 1E22-S004 has Excessive Voltage Drop, September 1, 1998.
2-00-06-090, Inadequate PMT Specified for Functional Check of Implementation of ECN 30253 of Remote Shutdown Panel, June 16, 2000.

Drawings

Schematic Diagram E02-1HP99, High Pressure Core Spray System (HP) HPCS Power Supply System (1E22-1070), Sheet 109, Revision R.
Schematic Diagram E02-1AP99, 4160 Bus 1A1 Diesel Feed Breaker 252-DG1KA, Sheet 013, Revision AC.
Internal External Wiring Diagram E03-1C61-P001, Remote Shutdown Panel 1C61-P001E, Sheet 10, Revision AD.
Internal External Wiring Diagram E03-1C61-P001, Remote Shutdown Panel 1C61-P001E, Sheet 12, Revision R.

Modifications

CC-21, Install manual flow control valve and flow indicator in FCHX outlet line, July 1997.
ECN 30855, Install Vacuum Breakers on 1VX06CC Chiller Condenser, November 19, 1999.
ECN 31570, Install Extension Pipe on Vacuum Breaker Line, May 24, 1999.
ECN 30364, Replace 1PSDG062C and 1PSDG063C, July 19, 1999.

ECN 31957, Use of pump motor replacement for Shutdown Service Water Pump, January 28, 2000.
ECN 31064, Corrective Action for CR 1-98-09-017 to Re-Wire the Control Circuit for the Closing Coil of the Division III EDG Output Breaker, November 6, 1998.
ECN 30253, Add Isolation Contacts to the Wiring for MCR Hand Switch 1HS-DG214, August 1, 1997.
ECN 30660, (1/14/98), Provides electrical schematic and wiring diagram to change the tap settings of the transformer for the Division I battery charger 1DC06E.
ECN 30445, (11/5/97), Provides corrective action for CR 1-97-10-241. Revises data sheet SC252 to reflect the new reset point for the High Voltage shutdown card in the associated battery charger.
ECN 31470, (8/19/99), Adds Voltage Transducer 1EY-SY051 in the SCAB Transformer 0AP01e control cabinet.
ECN 31776, (11/19/99), Provides new power source for Diesel Generator 1A ventilation instrumentation.
ECN 30708, (1/23/98), Replacement of the existing motor for the 1SX014B valve operator with a new motor.

10 CFR 50.59 Evaluations and Screenings

2000-0053 (Revision 0), CPS 3505.02, 12KV Distribution, Revision 12: Substation Modes of Operation.
2000-0050 (Revision 0), Turbine First Stage Pressure Sensing Protective Trip Functions During Bypass Valve Testing.
1999-150 (Revision 0), Change TS Bases SR 3.6.2.1.1 for Suppression Pool Temperature Limit of 95 degrees to “Nominal.”
1999-219 (Revision 0), SX Pump Motor Replacement and USAR Change Package 9-040.
1999-042 (Revision 0), SX Valve Control Circuit Changes.
1998-109, “ECN 30855 Shutdown Service Water Vacuum Breaker,” November 13, 1998.
1999-086, “ECN 31571 Vacuum Breaker Spray Deflector,” May 17, 1999.
1999-042, “ECN 31406 Shutdown Service Water Valve Control Circuit Changes,” March 8, 1999.
1999-127, Technical Specification Bases Change BL-98-004, June 12, 1999.
1999-219, ECN 31957 and UFSAR change package 9-040, January 28, 2000.
1999-075, USAR Change 8-294, April 22, 1999.
CPS 1005.06 F001, Engineering Evaluation of Minimum Voltage for Contractor1AP77E13A, October 18, 1999.
CPS 1005.06 F001, PDR 00-0146, CPS 8501.50 (R/I) Division II DG 4.16KV Protective Relay Functional Test, March 30, 2000.
Engineering Evaluation of Minimum Available Voltage for MCC Contractor 1AP77E13A, October 18, 1999.
PDR 00-0111, CPS 3506.01, Diesel Generator and Support System (DG), Revision 26b Revision 1a to CPS Diesel Generator Overspeed Trip Test, December 14, 1999.
CPS 3505.02 12KV Distribution (Revision 12).
CPS 1005.06F001, PDR 00-0132 to CPS 3501.01 CPS Procedure 3501.01 does not state a voltage match value tolerance between the UAT and RAT prior to transferring.
CPS 1005.06F001, PDR 2000-0133 to Revision 45 of CPS 9080.01 Changes to CPS 9080.01, Diesel Generator 1A (B) Operability - Manual and Quick Start Operability.
CPS 1005.06F001, 9433.13, Revision 32, ECCS Reactor Steam Dome Pressure B21-N097A(B) Channel Calibration PDR 00-0003.

CPS 1005.06F001, CPS 8507.01 (R/7) Division I/II Diesel Generator Maintenance.
CPS 1005.06F001, SX Pump Motor Replacement and USAR Change Package 9-040.
CPS 1005.06F001, Revision 1a to CPS 9080.31.
CPS 1005.06F001, Install Current and Potential Transformers for MPT 345KV Metering.
USAR Change 8-294, Update of Diesel Generator Loading and Use of 2000-hour Diesel Generator Rating, Revision 0.

Work Requests

AR No. D80263, Rewire the Control Circuit for the Division III EDG Output Breaker to Resolve the Excessive Voltage Drop Concern, June 14, 1998.
AR F11444, PMT for Division III Breaker Wiring, June 15, 2000

Miscellaneous

50.59 Performance Report January 2000, January 26, 2000.
50.59 Performance Report April 2000, April 28, 2000.
Engineering Quality Report - March 2000 Assessment No 2000-025, March 28, 2000.
NSED Self-Assessment Report Assessment No. 1999-134, August 26, 1999.
NEED Self-Assessment Report Assessment No. 2000-056, June 8, 2000.
Surv. 9080.02, Diesel Gen. 1C Operability - Manual and Quick Start Operability, January 15, 1999.
Engineering Evaluation 2-00-06-084, Evaluation of CR Immediate Corrective Action, June 22, 2000.
Self-Assessment Report 2000-026, Review of Post-Modification Testing, March 20, 2000.

Initial Document Request to licensee

I. Information Needed for In-Office Preparation Week

The following information is needed by Thursday, June 1, 2000, or sooner, to facilitate the selection of items to be reviewed during the onsite inspection week (June 12-16, 2000). The inspectors will select specific items from the information requested below and submit the selected items to your staff the week before the onsite inspection.

We request that the specific items selected from the lists be available and ready for review on the first day of inspection (June 12, 2000).

a. Permanent Plant Modifications

(1) List of permanent plant modifications to risk significant SSCs involving: (A) permanent plant changes; (B) design changes; (C) set point changes; (D) procedure changes; (E) equivalency evaluations; (F) suitability analyses; (G) calculations; (H) commercial grade dedications. [1]

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[1] Provide information requested going back two years.

(2) List of condition reports (open and closed) issued to address plant permanent modification issues/concerns. [1]

(3) Copy of modification procedure(s) and post-modification testing procedure.

b. Changes, Tests, or Experiments

(1) List of all 10 CFR 50.59 completed evaluations involving: (A) changes to facility (modifications); (B) procedure revisions; (C) tests or non-routine operating configurations; (D) changes to the USFAR; (E) calculation. [1]

(2) List of all 10 CFR 50.59 screenings that have been screened out as not requiring a full evaluation involving: (A) changes to facility (modifications); (B) procedure revisions; (C) tests or non-routine operating configurations; (D) changes to the USFAR; (E) calculations. [1]

(3) List of condition reports generated because of problems associated with 10 CFR 50.59 evaluations. [1]

(4) Copies of procedures that specify how 10 CFR 50.59 evaluations and screenings are performed.

(5) Copies of procedures that delineate how 10 CFR 50.59 FSAR updates are prepared by engineers or staff and how the licensee submits 10 CFR 50.59 FSAR updates.

(6) List of special tests or experiments and nonroutine operating configurations in the last two years (if any).

c. General Information

(1) List of procedure changes. [1]

(2) List of calculation revisions. [1]

(3) List of setpoint changes. [1]

(4) List of equivalency evaluations. [1]

(5) List of suitability analyses. [1]

(6) List of commercial grade dedications. [1]

7. List of Temporary Modifications

8. Latest Engineering Organization Chart

[1] Provide information requested going back two years

NOTE: If you have any questions regarding the requested information please contact Zelig Falevits at NRC Region III, Phone # 630-829-9717.