December 29, 2000

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: R. E. GINNA - NRC FIRE PROTECTION INSPECTION REPORT NO. 05000244/2000-009

Dear Dr. Mecredy:

This letter forwards the results of a triennial fire protection team inspection conducted on November 13-17, 2000, at the R. E. Ginna Nuclear Power Station. The preliminary results of the inspection were discussed with Messrs. J. Smith, Maintenance Superintendent, and R. Popp, Production Superintendent, and other members of your staff at an exit meeting on November 17, 2000.

This inspection was an examination of 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 purpose of the inspection was to evaluate your post-fire safe shutdown capability and fire protection program. Within these areas, the inspection consisted of selected examinations of procedures and representative records, observations of activities, and interviews with personnel.

No findings of significance were identified.

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 Daniel H. Dorman for/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000244
License No. DPR-18

Enclosure: NRC Inspection Report 05000244/2000-009

Attachments:
(1) Supplemental Information
(2) List of Documents Reviewed
(3) List of Acronyms Used
(4) NRC's Revised Reactor Oversight Process

cc w/encl:
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C. Donaldson, Esquire, State of New York, Department of Law
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F. William Valentino, President, New York State Energy Research and Development Authority
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DATE	12/20/00		12/20/00		12/20/00		12/29/00		12/18/00	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No: 05000244

License Nos: DPR-18

Report No: 05000244/2000-009

Licensee: Rochester Gas and Electric Company

Facility: R. E. Ginna Nuclear Power Plant

Location: 1503 Lake Road
Ontario, NY 14519

Dates: November 13 - 17, 2000

Inspectors: L. Scholl, Senior Reactor Inspector, DRS, Team Leader
R. Fuhrmeister, Senior Reactor Inspector, DRS
T. Walker, Senior Reactor Inspector, DRS
C. Cahill, Reactor Inspector, DRS
K. Young, Reactor Inspector, DRS

Approved By: W. H. Ruland, Chief
Electrical Branch
Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000244/00-09; on 11/13/00 - 11/17/00; Rochester Gas and Electric Company; R. E. Ginna Nuclear Power Plant; Fire Protection.

This inspection was conducted by a team of regional specialist inspectors. One licensee identified non-cited violation (NCV) was evaluated using the Significance Determination Process and found to be of very low safety significance (Green). The significance of issues evaluated under this process is indicated by their color (green, white, yellow, red). Corrective actions taken by the licensee appear reasonable. The NCV is listed in section 4OA7 of this report.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems

1R05 Fire Protection

This report presents the results of a triennial fire protection team inspection conducted in accordance with NRC Inspection Procedure (IP) 71111.05, "Fire Protection." The objective of the inspection was to assess whether RG&E has implemented an adequate fire protection program and that post-fire safe shutdown capabilities have been established and are being properly maintained. The screen house (SH), the auxiliary building basement and mezzanine (ABBM) and relay room (RR) fire areas were selected for detailed review. Due to the issuance of Change Notice 00-020 against Inspection Procedure 71111.05, "Fire Protection," the team did not review associated circuit issues during this inspection. This change notice has suspended this review pending completion of an industry initiative in this area.

.1 Fire Area Boundaries and Barriers

a. Inspection Scope

The team reviewed the adequacy of the design, installation and maintenance of the fire area boundaries including fire doors and fire dampers for the selected fire areas. The team also selected three fire barrier penetration seals for detailed inspection, including a review of the associated design drawings and qualification test reports. The team compared the observed in-situ seal configurations to the design drawings and tested configurations and ensured the seals were consistent with the ratings of the barriers in which they were installed. The design and qualification testing for raceway fire barriers were also reviewed and a walk down of installed barriers was performed for the selected areas.

b. Findings

Raceway Protection

10 CFR 50.48(b) states that for nuclear power plants licensed to operate prior to January 1, 1979, fire barriers installed to protect the post-fire safe shutdown capability, whether installed before or after 10 CFR 50 Appendix R became effective, are required to satisfy the technical requirements of Section III.G of 10 CFR 50 Appendix R. This section requires, in part, that raceway enclosures utilized in areas with fire detectors and automatic suppression shall have a one-hour fire rating. Generic Letter (GL) 86-10, "Implementation of Fire Protection Requirements," presents acceptable methods for satisfying the technical requirements of Appendix R, including guidance for fire barriers. GL 86-10 states that the documentation establishing the rating of a fire barrier should include the design description of the barrier and the test reports that verify its rating. The GL also provides the fire test acceptance criteria for establishing the fire rating of a barrier.

At the Ginna Station, Hemyc wrap has been utilized to protect selected raceways to meet the separation requirements of 10 CFR 50, Appendix R. The Hemyc design is typically a 1½-inch nominal alumina silicate fiber (a Kaowool type material) blanket encapsulated in Siltemp 84 fabric on the outside (side exposed to the fire) and a Klevers 600/6 fiberglass mat on the inside. Within the areas selected for inspection, Hemyc was installed to protect cable tray 111, conduit L398 and conduit L400 in the ABBM fire area. Since the Hemyc fire barriers were installed after the effective date of Appendix R, they are required to meet the technical requirements of Appendix R or have appropriate documentation to justify a deviation.

The team reviewed the following qualification test documentation utilized by RG&E as the bases for qualifying the Hemyc system as a one hour fire barrier for raceways:

- CTP-1026, “Fire Qualification Test of ‘Hemyc’ Cable Wrap System-One Hour,” June 1, 1982, Central Nuclear de Asco, Tarragona, Spain

- CTP-1077, “One Hour Fire Test on 3" Conduit for NES,” March 10, 1986, Southwest Research Institute, San Antonio, Texas

Test Report CTP 1026 provides the results of three tests designed to establish the one hour rating for raceway barriers. The team reviewed testing for the configurations that were most representative of the Ginna plant installations inspected by the team.

Test No. 1 included two 12-inch ladder-back cable trays stacked with eight inches of air gap between the trays. The top tray had 100% cable fill consisting of equal fills of 300 MCM cables, seven conductor #12 American Wire Gage (AWG) cables, and two conductor #16 AWG cables. The lower cable tray had a single layer of cables of the same ratio and types as in the top tray. The cable trays were enclosed in a single fire barrier enclosure consisting of an inner fiberglass cloth wrap, a single layer of 1½-inch Kaowool blanket installed over a steel framework that provided a minimum 2-inch air gap between the wrap and the trays and an outer layer of Siltemp fabric.

Test No. 2 consisted of a single 12-inch ladder-back cable tray with a single layer of cabling consisting of equal volumes of the same type cables utilized in Test No. 1. This cable tray was wrapped in the same manner as in Test No. 1.

Test No. 3 included a cable tray installation similar to that in Test No. 2 except with a 100% cable fill. This test also included a single 4-inch conduit with 100% cable fill. The conduit fire barrier system consisted of a 2-inch Cerablanket (a Kaowool type material) wrap covered by Siltemp wrap, held in place with ¾-inch finger straps.

These tests were conducted in accordance with ANI/MAERP “Standard Fire Endurance Test Method to Qualify a Protective Envelope for Class 1E Electrical Circuits.” NRC Generic Letter 86-10 specified ASTM E-119, “Standard Test Methods for Fire Tests of

Building Construction and Materials," as the acceptable testing method. Testing to the insurance standard rather than to ASTM E-119 presented the following concerns:

- ASTM E-119 is specific as to the type of thermocouple to be used in the testing, specifically fusion welded #18 AWG Chromel-Alumel wire. Test Report CTP 1026 states that "Pt., Pt-Rd" thermocouples were used which could affect the time delay and accuracy of the measurement used to determine the furnace temperature profile.

- ASTM E-119 requires a minimum of nine thermocouples distributed throughout the furnace to control the test fire curve. The thermocouples must also be mounted in minimum 12-inch long protective tubing and distributed throughout the furnace. This practice is necessary to confirm that the furnace has a uniform heat flux and is exposing all faces of the test article. Test CTP 1026 used only one thermocouple and one "reserve" thermocouple, both of which were located in the back of the furnace to control the test curve.

- ASTM E-119 requires that the minimum furnace be sized to test a 100 ft^2 sample. The furnace used in the test was only approximately 33 ft^2.

- ASTM E-119 requires thermocouples to be located on the cold-side surface of the barrier to measure at least nine separate points with an acceptance criteria that the temperature rise does not exceed 250 ^{o}F over the ambient starting temperature. The team found that the number and location of thermocouples may not have been sufficient to obtain an accurate thermal profile of the raceways. Specifically, Test Nos. 1, 2, and 3 utilized six, nine, and ten thermocouples, respectively and the test report did not state specifically where the thermocouples were located. Sketches included in the report indicated the thermocouples for Test No. 1 were located on the inside of the cable trays attached to cables with one thermocouple buried in the 100% filled tray and Test No. 2 had three thermocouples installed on cables located in the bottom of the tray and three located on the top of the tray.

The team identified several concerns regarding the measured temperatures. For example, during Test No. 1, all six recorded temperature rises were within that allowed by ASTM E-119. However, since the test configuration consisted of two cable trays inside the fire barrier system, including one with heavy (100%) fill which provided a significant thermal heat sink, the results may not bound the single, less than fully filled tray configuration reviewed by the team in the Ginna plant. Also, during Test No. 2, the temperatures of the thermocouples on the bottom of the tray exceeded the average allowable temperature rise at approximately 38 minutes and the temperature plot for the three thermocouples located on cables in the top of the tray stopped at 55 minutes, corresponding to a temperature rise of 250 ^{o}F. During Test No. 3, the single 4-inch conduit exceeded the allowable temperature rise at approximately 42 minutes and readings for all of the thermocouples in the 12-inch cable tray remained below the maximum allowable temperature.

Finally, the team noted that the potential effects of raceway supports on the test results appeared to be beyond the scope of all three of the tests, as the report does not discuss their interface or protection requirements. There were no thermocouples on or near supports to determine their impact on the temperature rise inside the fire barrier systems.

The team also reviewed Test Report CTP 1077 which was a one-hour test on a three-inch conduit. The team noted that the report introduction states that, “This report consists of an analysis of the one-hour fire exposure test performed on October 29, 1984, as an engineering test during the performance of a standard three-hour fire test on a different system. It is fully realized that this test is an engineering test only, and it does not qualify as a fully acceptable one-hour test. The purpose of this report is to analyze the data collected, determine the weak points, and recommend configurations for future evaluations.”

The test specimen consisted of a three-inch conduit with two 300 MCM cables, four #12 AWG seven conductor and eleven #16 AWG two conductor poly vinyl chloride (PVC) jacketed cables resulting in a 50% fill. The team identified similar concerns to those discussed above, with this test method as compared to ASTM E-119, including:

- The analysis does not provide sufficient information to determine the type of thermocouples that were used in the test.

- The test used only six thermocouples to measure internal barrier temperatures, of which, one appears to have failed at approximately four minutes into the test.

- The furnace used in the test was not described sufficiently to evaluate compliance with the standard.

- The test report does not state exactly where the thermocouples were attached to the test specimen.

The team noted that the Hemyc installations inspected at Ginna were in good condition and included protection of raceway supports. However, based on the review of test reports CTP 1026 and CTP 1077, the team determined that the results of the engineering testing alone were inconclusive for qualifying the fire barrier system as a one-hour rated fire barrier. Additionally, testing large conduits (3 and 4-inch) does not necessarily bound smaller sizes. Other barrier testing has demonstrated that small conduits may heat up more rapidly and reach higher end temperatures than larger conduits with the same fire barrier system. The Ginna installations inspected included 1½ and 2½-inch conduits.

The NRC has previously identified similar Hemyc qualification test issues at the Shearon Harris Nuclear Power Plant (IR 50-400/99-13) and the NRC Region II office requested the Office of Nuclear Reactor Regulation (NRR) assistance in Task Interface Agreement (TIA) 99-028, dated November 23, 1999, in evaluating the resolution of these items. This issue is unresolved pending further NRC review to determine whether the qualification tests of the Hemyc fire barrier wrap systems are acceptable. **(UNR**

05000244/2000-009-01) RG&E previously initiated commitment tracking item CAT 10050 to track NRC and industry developments associated with Hemyc fire barrier systems.

.2 Post-Fire Safe Shutdown Lighting and Communications

a. Inspection Scope

The team observed the placement and aim of emergency battery lights (EBLs) throughout the selected fire areas to evaluate their adequacy for illuminating access and egress pathways and any equipment requiring local operation for post-fire safe-shutdown. The team also reviewed preventative maintenance procedures and documents to determine if adequate surveillance testing was in place to ensure operation of the EBLs. These procedures and documents included eight-hour discharge testing of EBLs, EBL battery replacement, and monthly inspection and six-month inspection of Appendix R EBLs. The team reviewed Ginna Station's communication plan, surveillance procedures conducted on the communication systems, and maintenance performed on the portable radio battery system to determine if communications could be maintained in the event of a fire at the site.

b. Findings

No findings of significance were identified.

.3 Programmatic Controls

a. Inspection Scope

During tours of the facility, the team observed the material condition of fire protection systems and equipment, the storage of permanent and transient combustible materials, and the control of ignition sources. These reviews were performed to evaluate the implementation of fire protection program administrative controls. The team also reviewed design control procedures to verify that plant changes were adequately reviewed to assess the potential impact on the fire protection program and the safe shutdown equipment and procedures.

b. Findings

No findings of significance were identified.

.4 Fire Detection and Suppression Systems and Equipment

a. Inspection Scope

The team reviewed the design, installation and maintenance of selected manual fire suppression equipment including standpipe systems, manually actuated fixed suppression systems, and portable extinguishers. The team also inspected the fire brigade's protective ensembles for structural fire fighting and portable communications equipment. Automatically actuated fixed fire suppression systems were also reviewed. This review included hydraulic calculations, discharge and functional tests and field walk downs of selected systems including the Halon fire suppression systems for the relay room and the automatic sprinkler systems in the screen house and auxiliary building.

b. Findings

No findings of significance were identified.

.5 Alternative Shutdown Capability

a. Inspection Scope

The team reviewed the Ginna Station Safe Shutdown Analysis, emergency response procedures and fire response procedures for the selected fire areas to evaluate the methods and equipment used to achieve hot shutdown following postulated fires. The team reviewed piping and instrumentation drawings for post-fire shutdown systems to determine required components for establishing flow paths, to identify equipment required to isolate flow diversion paths, and to verify appropriate components were on the safe shutdown equipment list. The team also reviewed selected alternate shutdown components and their control circuits to ensure that proper isolation was provided for alternate shutdown capability and performed field walk-downs to evaluate the protection of the equipment from the effects of fires.

Post-fire shutdown procedures for the selected areas were also reviewed to determine if appropriate information was provided to plant operators to identify protected equipment and instrumentation and if recovery actions specified in post-fire shutdown procedures considered manpower needs for performing restorations and area accessibility. The team also reviewed training lesson plans and job performance measures for the alternative shutdown procedures, discussed training with licensed operators, reviewed selected alternate shutdown transfer switch tests, reviewed minimum shift manning required by technical specifications, and evaluated the accessibility of the alternative shutdown operating stations and the accessibility of required manual action locations.

b. Findings

No findings of significance were identified.

.6 Safe Shutdown Circuit Analyses

a. Inspection Scope

The team reviewed the Ginna Station Fire Protection Program Report (FPPR), dated October, 18, 2000, and design analysis DA-EE-2000-066, “Appendix R Conformance Analysis,” Revision 0, to assess the adequacy of the methodology applied in the analysis. The team also reviewed assumptions utilized in the analysis, the application of NRC guidance and the adequacy of engineering evaluations of design vulnerabilities.

The team reviewed power and control cable routing for a sample of components required for post-fire safe shutdown to determine if the cables were properly routed outside the fire area of concern or protected against the effects of fire. The adequacy of raceway fire barriers is discussed in section 1RO5.1 of this report. The team also reviewed selected control circuit schematic drawings to determine if the licensee had properly evaluated the circuits for the effects of hot shorts, open circuits and shorts to ground.

The team reviewed electrical fuse and circuit breaker coordination studies to ensure equipment needed to conduct post-fire safe shutdown activities would not be impacted due to a lack of coordination. The team also reviewed the electrical isolation capability of selected equipment needed for post-fire safe shutdown to ensure that such equipment could be operated locally, if needed.

Due to the issuance of Change Notice 00-020 against Inspection Procedure 71111.05, "Fire Protection," the team did not review associated circuit issues during this inspection. This change notice has suspended this review pending completion of an industry initiative in this area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Corrective Actions for Fire Protection Deficiencies

a. Inspection Scope

The team reviewed the fire impairments log, open corrective maintenance work orders for fire protection and safe shutdown equipment, selected action requests for fire protection and safe shutdown issues to evaluate the prioritization for resolving fire protection related deficiencies and the effectiveness of corrective actions. The team also reviewed recent Quality Assurance Surveillance and Audit Reports, and Engineering Self-Assessments of the fire protection program to determine if RG&E was identifying program deficiencies and implementing appropriate corrective actions.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) LER 50-244/2000-002 Assumed Plant Fire May Cause Loss of Emergency Diesel Generator

RG&E identified that a fire in the screen house could result in short circuits in the control cable associated with the under voltage protection relays for safeguards bus 17 and the "B" emergency diesel generator (EDG). RG&E determined that the short circuits could result in blown control power fuses for the "B" EDG resulting in the loss of the EDG. The safe shutdown analysis for a postulated fire in the screen house assumes that the "A" EDG would be lost as a result of fire damage. The loss of both EDGs due to a screen house fire is a condition that is outside the design basis of the plant.

To resolve this issue RG&E installed a plant modification to add additional fuses that coordinate with the upstream EDG control circuit fuses. Specifically, in the event of fire damage to the affected cable, the added fuses would open to isolate the damaged circuit before the EDG control power fuses blow.

The team discussed this event with RG&E personnel and reviewed associated electrical drawings and circuit breaker coordination study associated with bus 17 to determine whether any additional failures caused by a fire in the screen house could result in the loss of the offsite power to the remaining safety buses necessary to shut down the plant. Based on this review, the team concluded that offsite power would remain available. This item was screened using the Significance Determination Process and found to be of very low safety significance (Green) since offsite power remains available to power equipment necessary for post-fire safe shutdown.

However, the failure to isolate all potentially affected cables within the screen house constitutes a violation of the requirement of 10 CFR 50, Appendix R, Section III.G.3 that requires the alternative shutdown capability be independent of the area of concern. Since this issue was identified by the licensee, found to be of very low safety significance, and has already been corrected, the violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 05000244/2000-009-02)**

4OA6 Meetings, Including Exit

The inspectors presented their preliminary inspection results to Messrs. J. Smith, R. Popp, and other members of your staff at an exit meeting on November 17, 2000.

4OA7 Licensee Identified Violations. The following finding of very low significance was identified by RG&E and was a violation of NRC requirements which met the criteria of Section VI.A of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation.

NCV 05000244/2000-009-02 10 CFR 50, Appendix R, Section III.G.3 requires that the alternative shutdown capability be independent of the fire area of concern. RG&E identified a cable in the screen house that could result in the loss of both EDGs due to a fire in that area (section 4OA3.1). Reference Action Report 2000-1146 and LER 2000-02.

If RG&E contests this NCV, or its significance, RG&E should provide a response within 30 days of the date of this inspection report, with the basis for the denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the R. E. Ginna facility.

ATTACHMENT 1 - SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Rochester Gas and Electric

R. Biedenbach, Manager, Safety Services
D. Dean, Control Room Foreman
G. Graus, Manager, I&C/Electrical Engineering
E. Groh, Control Room Foreman
D. Gomez, Shift Supervisor
B. Hunn, System Team Engineer
T. Joachimczyk, System Team Engineer
R. Ploof, Manger, Balance of Plant Systems
P. Sidelinger, EOP Procedures Coordinator
F. Machiuska, Manager, Operations Training
P. Swift, System Team Engineer
C. Vitali, Fire Protection Engineer
D. Wilson, Licensing Engineer
G. Wrobel, Nuclear Safety and Licensing
B. Zollner, Senior License Instructor

Nuclear Regulatory Commission

H. Nieh, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

UNR 05000244/2000-009-01	Adequacy of Hemyc Cable Wrap Fire Barrier Qualification Test and Evaluation

Opened and Closed

NCV 05000244/2000-009-02	Failure to ensure that the alternate shutdown capability for the screen house was electrically and physically independent of the area

Closed

LER 05000244/2000-002	LER 50-244/2000-002 Assumed Plant Fire May Cause Loss of Emergency Diesel Generator

ATTACHMENT 2 - LIST OF DOCUMENTS REVIEWED

Piping and Instrumentation Drawings

P&ID 33013-1250, “Station Service Cooling Water Safety Related (SW),” Rev. 32
P&ID 33013-1238, “Standby Auxiliary Feedwater (FW),” Rev. 19
P&ID 33013-1237, “Auxiliary Feedwater (FW),” Rev. 40
P&ID 33013-1265, “Chemical Volume and Control System Charging (CVCS),” Rev. 8
P&ID 33013-1607, “Fire Protection System Yard Loop,” Rev. 22

Control Circuit Schematics / Wiring Drawings

03201-0102, Rev. 10, 120 V AC Instrument Bus
03202-0102, Rev. 10, 125 V DC Power Distribution System
03202-0652, Rev. 19, 480 Volt
10905-0270, Rev. 3, Elementary Wiring Diagram Control Valve Table
10905-0295, Rev. 2, Elementary Diagram Remote Operated Valves

21946-0071A, Sh. 1, Rev. 3	480 V Bus 14 - Unit 23B Charging Pump A PCH01A(52/CHP1A) Sh. 2, Rev. 1
21946-0071B, Sh. 1, Rev. 3	480 V Bus 16 - Unit 15B Charging Pump B PCH01B(52/CHP1B) Sh. 2, Rev. 1
21946-0239, Sh. 1, Rev. 6	Turbine Driven Aux. FW PMP DC Lube Oil PMP PL011 (72/TAOP) Sh. 2, Rev. 2
21946-0446, Sh. 1, Rev. 5	480 V Bus 16 - Unit MC Standby Aux. Feedwater Pump D PSF01B (52/SAFWP1D) Sh. 2, Rev. 1

21946-0650, Rev. 7, TDAFW Pump Steam Supply VLV MOV-3504A 3504A (42/3504A)
21946-0651, Rev. 5, TDAFW Pump Steam Supply Valve MOV-3505A 3505A (42/3505A)
21946-0656, Rev. 6, TDAFW Pump Discharge VLV MOV-3996 3996 (42/3996)
33013-0652, Rev. 19, 480 Volt One Line Wiring Diagram
33013-1736, Sh. 1, Rev. 14, Diesel Generator A Control Schematic, Sh. 2, Rev. 14
33013-2539, Rev. 5, AC System Plant Load Distribution

Calculations/Design Documents

DA-EE-93-107-07, Rev. 2,	480 Volt Coordination and Circuit Protection Study
DA-EE-93-107-07, Rev. 3,	4160 Volt Over current Relays Coordination and Circuit Protection Study
DA-EE-99-066, Rev. 2,	DC System Fuse Coordination Ginna Station TSR 97-190
DA-EE-2000-066, Rev. 0,	Appendix R Conformance Analysis Ginna Station
DA-ME-94-004, Rev. 0,	Design Analysis Ginna Station Fire Protection Hose Reels
DA-ME-96-070, Rev. 0,	Hydraulic Sprinkler Design Calculations for Fire Protection Systems Off the Diesel and Motor Driven Fire Pumps
DA-ME-2000-052,	Fire Detection System Evaluation
DA-ME-2000-075, Rev. 0,	Pressurizer, Volume Control Tank and RWST Evaluations for Appendix R
DV16B, EWR 3729, Rev. 1,	Relay Room Halon System
DV16W, Rev. 0, EWR 3729,	Relay Room Halon System
ME-199, Rev. 0, EWR 3729,	Relay Room Halon System
PENQ-11, Penetration # A-133-P	Release 9-81 for system S08, Ansul Hydraulic System Design Program

ATTACHMENT 2 - LIST OF DOCUMENTS REVIEWED (cont.)

Test No. 1078, dated 5/21/84, Halon 131 System Test Record
1-DC-785-0428-15, Screen House Sprinkler System Task 13
PCR# 95-038, Rev. 2, Fire Pump Diesel Replacement
PCR# 96-121, Rev. 0, Removal of Service Water Pump Remote Control Switches
PCR# 99-032, Rev. 0, S22 & S23 Detection Modification

Procedures

A-56, Communication Systems at Ginna Station, Rev. 25
AP-CR.1, Rev. 16, Control Room Inaccessibility
CME-44-02-MCC A/B, Rev. 2, Westinghouse Motor Control Center A Unit Molded Case Circuit Breaker Maintenance for MCC A/B
CME-50-02-52/CHP1B, Rev.0, Westinghouse, 480 V Air Circuit Breaker, Type DB-50, Charging Pump B, Bus 16, Position 52/CHP1B
CPI-APPX-R-SR-32, 3/3/00, Calibration of Appendix R Source Range, N32R
CPI-FI-115, 5/27/00, Calibration of Reactor Coolant Pump 1A Seal Injection Flow Indicator FI-115
CPI-FI-116, 5/5/99, Calibration of Reactor Coolant Pump 1B Seal Injection Flow Indicator
CPI-FLO-2015A, 7/14/99, Calibration of the Turbine Driven Auxiliary Feedwater Pump Discharge Flow Loop 2015A
CPI-LT-428A, 3/9/99, Calibration of Pressurizer Level Transmitter LT-428A
CPI-LT-460A, 3/12/99, Calibration of Steam Generator A Wide Range Level Transmitter LT-460A
CPI-LVL-428A, 5/9/97, Calibration of Pressurizer Level Loop 428A Rack Instrumentation
CPI-LVL-460A, 5/23/97, Calibration of Steam Generator A Wide Range Level Loop 460A Rack Instrumentation
CPI-LVL-2022A, 7/15/99,Calibration of Condensate Storage Tank Level Loop 2022A and Pressure Indicator PI-2809
CPI-PI-70.5, 11/19/99, Calibration of Plant Field Mounted Pressure Gauges
CPI-PRESS-420B, 4/4/97, Calibration of Reactor Coolant System Pressure Channel 420B Rack Instrumentation
CPI-PRESS-469A, 4/4/97, Calibration of Steam Generator A Steam Pressure Loop 469A Rack Instrumentation
CPI-PT-420B, 3/99, Calibration of Reactor Coolant System Pressure Transmitter PT-420B
CPI-PT-469A, 12/8/99, Calibration of Steam Generator A Steam Pressure Transmitter PT-469A
CPI-SPARE-DET-N32, 3/16/98, Operability Checkout of Source Range Spare Detector N32
CPI-TEMP-409B-2, 5/5/00, Calibration of Reactor Coolant Loop A Cold Leg Temp. 409B-2
CPI-TEMP-409A-2, 5/6/99, Calibration of Reactor Coolant Loop A Hot Leg Temperature 409A-2
EP-3-P-0132, Rev. 4, Fire Protection/Appendix R Conformance Verification
EP-3-S-306, Rev. 13, Change Impact Evaluation Form
ER-FIRE.1, Rev. 7, Alternate Shutdown for Control Complex Fire
ER-FIRE.3, Rev. 6, Alternate Shutdown for Aux Building Basement/Mezzanine Fire
ER-AFW.1, Rev. 20, Alternate Water Supply to the AFW Pumps
ER-D/G.2, Rev. 11, Alternate Cooling for Emergency D/Gs
FRP-5.0, Rev. 3, Auxiliary Building Intermediate Floor
FRP-19.0, Rev. 4, Relay Room/Multiplexer Room/Annex Room
FRP-30.0, Rev. 2, Screen House Basement
FRP-31.0, Rev. 2, Screen House Operating Floor

GME-45-99-01, Rev. 4, Electric Motor Inspection and Maintenance
GME-50-02-DB50, Rev. 13, Westinghouse, 480V Air Circuit Breaker, Type DB50 Maintenance for Type DB-50 Breakers
IP-DES-2, Rev. 11, Plant Change Process
IP-DES-3, Rev. 4, Temporary Modifications
IP-PRO-4, Rev. 3, Procedure Adherence Requirements
M-32-6, Rev. 9, Molded Case Circuit Breaker Bench Test Trip Test
ND-FPP, Rev. 5, Fire Protection Program
PCN 99-4124 for ER-FIRE.3
PM200000, Rev. 3, 8 Hour Discharge Test on Appendix R Emergency Lights
PM200002, Rev. 3, 6 Month PM Inspection on Specific Appendix R Emergency Lights
PM200004, Rev. 2, Appendix R Emergency Light Monthly Inspection
PM200015, Rev. 0, Appendix R EBL Battery Replacement
SC-3.15.15, Rev. 71, Emergency Fire Equipment Inventory and Inspection

Training Materials

JPM J064.001, Rev. 7, Lineup Alternate Cooling to EDG
JPM J039.001, Rev. 7, Locally Close MSIVs
JPM J004.009, Rev. 2, Take Local Manual Control of Charging Pump
JPM J064.004, Rev. 10, Start ‘A’ EDG Locally per ER-FIRE.1
JPM J061.008, Rev. 3, Open TDAFW Pump Steam Supply Valves
JPM J061.005, Rev. 5, Reset TDAFW Pump Turbine Trip/Throttle Valve
JPM J004.005, Rev. 1, Locally Realign Charging Pump Suction to RWST
Lesson Plan RSC56T, Rev. 4, Alternative Shutdown
Lesson Plan RER22C, Rev. 2, Appendix R Alternative Shutdown
Lesson Plan RSC62D, Rev. 3, Appendix R - Alternative Shutdown

Corrective Action Program Documents

AR-0340, Fire Suppression System Operability Concern
AR-98-1265, Fire Detection System Basis not Consistently Utilized
AR 99-0088, Feasibility of Performing Control Operator Actions in ER-FIRE.3
AR-99-0661, Z16D1X Failed Sensitivity Test
AR-99-1304, Spurious Fire Alarm on Z-26 Screen House
AR-99-1337, Diesel Fire Pump will not Start
AR-99-1486, A-52.12 Not Updated to Reflect All Systems Correctly
AR-2000-0182, Appendix R Hemyc Junction Collar
AR-2000-0603, City Yard Loop is not 100% Backup to Fire Service Water System
AR-2000-0647, Motor Driven Fire Pump Will Not Trip
AR-2000-1067, Couldn’t Fill Fire Water Booster Tank
AR 2000-1538, ER-FIRE.1 procedure not in accordance with UFSAR and Appendix R report
CAI Tracking # 8 Fire Pump Test Line Requires Flow Set point Basis
CAI Tracking # 9, Fire Pump Test Line Requires Pressure Set point Basis

Drawings

21488-100, Sh. 3, 4, 6, Fire Smoke and Pressure Barriers

ATTACHMENT 2 - LIST OF DOCUMENTS REVIEWED (cont.)

21488-0204, Sh. 1, Rev. 1	Hemyc Fire Wrap, Aux. Bldg. App. R Circuits C687, E53, L398, L400; Appendix R Cable Tray 111
Sh. 3, Rev. 1	Hemyc Fire Wrap, Aux. Bldg. App. R Circuit L400 and C687
Sh. 7, Rev. 1	Hemyc Fire Wrap, Aux. Bldg. App. R Circuit L398
Sh. 8, Rev. 1	Hemyc Fire Wrap, Aux. Bldg. App. R Circuit L398

Safety Evaluations

SE 2270 for Procedure 2.4.143.1, Revision 0, 12/22/87
SE 2271 for Procedure 2.4.143.2, Revision 0, 12/22/87
Preliminary Evaluation Checklist for Procedure 2.4.143.1, Revision 8

Quality Assurance Documents

AINT-1998-0009-RTD, Fire Protection and Safety Audit Report, 02/22/99
AINT-2000-0012-TJD, Fire and Safety Audit, 08/15/00
SQUA-1999-0008-JMT, Plant Shutdown From Hot Shutdown to Cold Conditions
SQUA-1996-0072-NAB, Engineering Activities During GSU Replacement Outage
SQUA-1996-0016-TGT, Mechanical Maintenance Activities
SQUA-1996-0064-CER, Falsification of Records - Maintenance and Plant Support
SQUA-1996-0065-JJF, Appendix R Reviews in PCRs
SQUA-1995-0004-GFS, Walkdown surveillance of fire barrier seals
SQUA-1995-0036-TGT, EWR 4812 - 5A/5B Heater Replacement
SQUA-1999-0007-JMT, 1999 Outage Shift Turnovers

Other Documents

RG&E Stationary Battery & Charger Inspection Report (10/21/99, 4/24/00, 10/17/00)
Vendor Manual #D1140-0377, Dual Lite, RG&E Emergency Lighting
System Walkdown checklist dated 10/6/2000
CATs ID 10319, Review of IN 2000-12, “Potential Degradation of Fire Fighting Primary Protective Garments”
Fire Brigade Drill Critique Form SC-3.4.1:7, dated 10/29/00, 11/4/00 and 11/11/00

ATTACHMENT 2 - LIST OF DOCUMENTS REVIEWED (cont.)

Work Orders

WO 19603003, Rev. 1	Perform Electrical Tests on MO/CHP1B, Perform a 3 minute Megger Test, DC Hi Pot, and a DC Surge Test on Charging Pump 1B
WO 19900586, Rev. 1	Perform PM on 52/CHP1B per Procedure GME-50-02-DB50
WO 19902455, Rev. 1	Perform PM Inspection on 52/CHP1A Breaker Charging Pump 1A Breaker - Bus 14/23B Primary Injection Test is Required
WO 19902457, Rev. 1	Perform Electrical Tests on MO/CHP1A, Perform a 3 Minute Megger Test, DC Hi Pot, and a DC Surge Test on Charging Pump 1A Motor
WO2000117, Rev. 1	Adjust Amptector Set Points in accordance with Attachment 1 to PCR99-085

ATTACHMENT 3 - LIST OF ACRONYMS USED

ABBM	Auxiliary Building Basement and Mezzanine
AC	Alternating Current
ASTM	American Society of Testing Materials
AWG	American Wire Gage
DRS	Division of Reactor Safety
EBL	Emergency Battery Lights
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
^{o}F	Degrees Fahrenheit
FPPR	Fire Protection Program Report
ft^2	Square Feet
GL	Generic Letter
IN	Information Notice
JPM	Job Performance Measure
MCM	Thousand Circular Mils
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
P&ID	Piping and Instrumentation Drawing
PVC	Polyvinyl Chloride
RG&E	Rochester Gas and Electric
RR	Relay Room
SE	Safety Evaluation
SH	Screen House
TIA	Task Interface Agreement

ATTACHMENT 4 - 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.