U.S.NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-244/93-05

Docket No. 50-244

Licensee No. DPR-18

Licensee: Rochester Gas and Electric Corporation

Facility Name: Ginna Nuclear Power Station

Inspection At: Ontario, New York

Inspection Conducted: February 16-19, 1993

Inspectors: J.E. Carrasco — 03-25-93
J.E. Carrasco, Reactor Engineer, — date
Materials and Processes Section, EB,DRS

Approved by: for — 3/26/93
E. H. Gray, Chief, Materials — date
EB,DRS

Inspection Summary:

Areas Inspected: A safety inspection was conducted to determine whether the licensee's design provisions for modification EWR 5275, "Replacement of Containment Recirculating Fan Coolers (CRFCs)," were performed in accordance with regulatory requirements which are outlined in the FSAR; these are General Design Criteria 38, 39, and 40. In addition, modification EWR 5411, "Service Water Temporary Cooling," was inspected.

Results: The pertinent analysis to demonstrate that the replacement of containment recirculation fan coolers meets the General Design Criterion 38 was completed. Based on the licensee's preliminary assessment regarding the loads on the D/G resulting from the new CRFC, it appears that these loads would not adversely affect the D/G. The modification improved access to permit periodic inspections of the CRFC coils.

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Commitments (RG&E Nos. RO1968 and RO1969) to resolve weaknesses identified in NRC Inspection No. 91-201 Service Water System inspection were not properly followed up. This item is unresolved pending NRC verification of a letter describing the deferral of these two commitments to the 1994 outage.

1.0 PURPOSE AND SCOPE

The purpose of this inspection was to determine whether the licensee's design provisions for modification EWR 5275, "Replacement of Containment Recirculating Fan Coolers (CRFCs)," were in accordance with regulatory requirements outlined in the FSAR and General Design Criteria 38, 39, and 40. In addition, modification EWR 5411, "Service Water Temporary Cooling," was inspected.

2.0 REVIEW OF MODIFICATION EWR 5275 CONTAINMENT RECIRCULATION FAN COOLER (CRFC) COIL REPLACEMENT (37700)

The objective of the modification was to improve the overall reliability of the CRFCs by minimizing the occurrence of cooling coil tube leakage problems, improve the accessibility to the cooling coils for inspection, cleaning and repair, and provide additional heat transfer capability.

2.1 Existing System Configuration

At R. E. Ginna Nuclear Power Plant the containment recirculation, cooling, and filtration system was designed as Seismic Category I. At the present, the air recirculation system consists of four air handling units. Each unit included motor, fan, cooling coils, moisture separators and high efficiency particulate air (HEPA) filters, duct distribution system, and instrumentation and controls. The units are located on the intermediate floor of containment at elevation 253'-3" between the containment wall and the primary compartment shield walls.

Two of the four air handling units are equipped with activated charcoal filter units, normally isolated from the main air recirculation stream, through which the air-steam mixture may be bypassed to remove volatile iodine following an accident. During normal operation, the flow sequence through the air handling units is as follows: cooling coils, moisture eliminator, HEPA filters, fan, and discharge header. Two of the four air handling assemblies are required for the post-accident depressurization of the containment.

The CRFC coils were designed to remove heat losses from the reactor coolant system and other systems and equipment in the containment during normal plant operation. During a design basis main steam line break accident, the CRFC, in conjunction with the containment spray system, keeps the containment pressure from exceeding design pressure by removing energy from the containment via the service water system. In addition, the CRFC removes energy from the containment via the service water system following other design basis accidents, such as a loss of coolant accident.

R. E. Ginna Nuclear Power Plant has experienced an increasing number of CRFC coil tube leaks. These leaks have required containment entries during operation for inspection and repair with consequent personnel exposure. The licensee indicated that the leaks were detected in the vicinity of the casing end-plate and the tube stubs connected to the service water system inlet and outlet headers of the CRFC coils. Furthermore, the licensee determined that the cause of the observed leaks was attributed to erosion/corrosion of the copper tubing. Destructive examinations of removed tube sections show a grooved appearance typical of erosion on the side of the tube in the vicinity of the return bends. It was noted that the average tube water-side velocity during normal operating conditions was in excess of 10 feet per second (fps). Industry experience and recommendations of industry standards groups were to maintain water velocities to less than 6 fps for copper tubing to avoid erosion.

The licensee evaluated the design, performance characteristics and basis for the existing CRFC cooling coils. As a result of this evaluation, the licensee decided to replace the twelve (three per unit) existing cooling coils with coils of a new design. This replacement was performed via Engineering Work Request (EWR) 5275.

2.2 Findings

Based on the review of the modification against general design criterion 38 of 10 CFR 50 Appendix A, the inspector found that this modification replaced the CRFC Cooling Coils with coils of enhanced design. The heat removal capability of the enhanced design had been specified, via purchase specification, with margin above the accident analysis minimum requirement and by both conservative and best estimated analytical techniques. The actual design of the coils is expected to perform at higher heat removal rates than that required by the purchase specification.

The inspector reviewed the impact of the new CRFC on the containment design basis and noted that the upper-bound limit for maximum heat removal capability was estimated to exceed the current limit stated in the licensee's UFSAR. The maximum heat removal limit is set by the containment backpressure assumed by the fuel vendor for design basis, large break loss of coolant accident analysis of peak clad temperature (PCT). Lower containment backpressure results in higher PCT. The licensee intends to use the new, higher estimated maximum heat removal of the new coolers to reevaluate containment backpressure, although less than the current estimate, it is still greater than the backpressure assumed by the fuel vendor for PCT analysis. Therefore, it is still within the current analytical limits.

Although the service water pump and the CRFC fan motor operating points may change slightly due to different flow resistances of the new cooler coils, the Diesel Generator (D/G) loads are not expected to be adversely affected since current D/G load estimates were based on peak motor power demands for both SW pumps and CRFC fan motors. The licensee indicated that this preliminary assessment would be confirmed and formally documented as part of the overall EWR design process.

The inspector verified that this modification only affected the CRFC coils, and access to other components of the system, including HEPA filters, fans, dampers, ductwork, and charcoal filters, were unaffected.

Air side access for periodic inspection of the coils is being maintained. However, the larger cooling coils will reduce the access inspection space between the coils and demister from 18" to 12". The licensee has indicated that 12" is considered sufficient space for periodic inspection. However, waterside access has been greatly enhanced by the addition of the cooler coil waterbox, which provides ready access to the tubeplate and tube-ends for maintenance, inspection, and repair/plugging of tubes. Such access was not provided with the existing cooling coils.

2.3 Conclusion

Based on the findings discussed above, the inspector concluded that the licensee performed an adequate thermo-hydraulic analysis and has improved access to permit periodic inspections of the CRFC coils.

2.4 Assessment of Integrity and Periodic Testing of CRFC System

The inspector found that the modification involved only minor piping changes. Specifically, service water (SW) piping adjacent to the coolers is being converted from welded to flanged connections to permit easier access to the coils via the waterbox. In the new coils, the flanged spools can be removed to accommodate removal of the waterbox cover plate. As a result, the ability to perform periodic pressure and functional testing to assure the structural and leaktight integrity of the coils is not adversely affected. Furthermore, the piping would remain under the ASME Code, Section XI, ISI program for periodic inspection and leak testing.

Operation of the associated cooling water system could potentially affect the overall function of the SW system. Specifically, the new coils have a reduced hydraulic resistance so that they will demand more flow from the SW System, potentially decreasing flow to other safety-related equipment cooled by Service Water. The licensee is performing analyses of the SW system to estimate the expected flow distribution to all safety-related SW coolers and heat exchangers to demonstrate that no adverse changes in flow delivery or pump performance are expected as a result of the CFRC coil modification. These evaluations would be confirmed as part of the post-modification testing of the new CRFCs.

The basic flow circuits for air and water are not significantly changed by this modification except that hydraulic resistance to both airside and waterside flows change slightly. The modification includes the addition of several new SW pressure gauges and airside duct transverses for air flow meters to enhance the ability to monitor and test for functional capability and operability.

As part of their commitment to Generic Letter 89-13, the licensee intends to perform periodic thermal performance tests and monitoring of the new CRFCs. The conditions for the test will be normal operating SW and containment atmospheric conditions which are not readily representative of the high temperature, high pressure steam/air mixture expected in containment during design basis conditions. However, the tests will attempt to properly account for differences in conditions and extrapolate results to predict performance at design basis conditions.

2.5 Conclusion

Based on the assessment and findings discussed above, the inspector concluded that the licensee's new CRFC replacement would permit periodic testing of the coils and associated components to assure system integrity and system operability.

3.0 REVIEW OF MODIFICATION EWR 5411 SERVICE WATER TEMPORARY COOLING

The licensee's modification EWR 5411 was prepared to design and install a temporary cooling water system (TCWS) to provide adequate cooling capability for critical SWS cooling demands during a SW system outage.

3.1 Existing System configuration

The licensee's SW system in its present configuration takes water from Lake Ontario via the screen house and supplies cooling water to various turbine plant loads as well as auxiliary reactor plant loads. The SW system supplies seal water to the recirculating water pumps and the vacuum pumps, flushing water to the traveling screens and makeup water to the fire water storage tank via the fire booster pump. Service water is also the normal supply to the standby auxiliary feedwater system and an alternate supply to the auxiliary feedwater system. The SW system was designed to provide adequate cooling to essential and nonessential loads during normal operations and to essential loads during accident conditions. The system discharges back into Lake Ontario via the discharge canal or via Deer Creek.

The SW system supplies cooling water via a loop supply header which splits into two headers with two service water pumps supplying each header (namely, headers "A" and "B"). Cross-connection valves are located in the supply loop to split the two service water headers.

The SW system piping and components have been exposed continuously to the environment of Lake Ontario for over 20 years. This environment is gradually diminishing the SW system ability to perform its intended functions. Many system valves are no longer leak tight and provide poor isolation.

The licensee initially conceptualized the service water temporary cooling modification to provide alternate cooling capability for critical plant cooling demands during the proposed service water system outage scheduled to coincide with the 1993 Ginna Station refueling outage and full core off load. The duration of the proposed service water system outage is currently estimated at 10-15 days. The licensee stated that this original design concept had been modified. The latest concept reflects a temporary cooling water system (TCWS) designed to support a Service Water System outage reduced in scope from a complete system out of service to only a single loop "B".

3.2 Findings

The inspector reviewed the conceptual design, safety analysis, pertinent P&IDs, design drawings, and related documentation for the TCWS ("B" Service Water header Outage). The TCWS, as the modification drawings indicated, consisted of a permanent portion and a temporary portion. The inspector focused his review on the temporary portion that consisted of hoses necessary to connect the permanent valve to the heat exchangers. Specifically, the TCWS design would serve the spent fuel pool heat exchanger "A," standby spent fuel pool heat exchanger, and diesel generator "B" jacket water heat exchanger and lube oil cooler.

The inspector verified that sufficient provisions were made to assure that adequate heat transfer capability of the spent fuel pool remained within design basis. The SFP heat load is less than 14 Million BTU/hr and the service water temperature is less than 60° F.

The "A" SW header was designated as Seismic Category I. This implied that it will remain functional during and after a seismic event. The standby heat exchanger is cooled by SW via a hose connection to the "A" SW header. The standby system is non-seismic. However, the licensee indicated that it is properly restrained to prevent any damage to safety-related equipment in the vicinity during a seismic event.

The inspector reviewed the service water system documentation. Specifically, Attachment A of a letter from the licensee (RG&E Corporation) to the USNRC's Director of the Division of Reactor Projects, I and II, dated April 6, 1992, which discussed the licensee's initial response to the USNRC's Inspection Report's Unresolved Item 91-201-09, "Weakness in Generic Letter 89-13 Implementation."

In Item five, the licensee stated that a SW system-specific erosion/corrosion inspection plan will be developed for the 1993 outage. In Item six, the licensee stated that robotic inspection technology will be used to inspect several points of access to the underground SW system piping in conjunction with major SW system maintenance efforts scheduled for the 1993 annual refueling outage. The inspector inquired in regard to the status of these commitments, and the licensee indicated that these two commitments were deferred to the 1994 outage. At the exit meeting, the inspector expressed a concern in regard to the accuracy of correspondence to the NRC regarding statements to inspect the SW system piping. The licensee acknowledged this concern. The inspector stated that a letter to the originator (NRR) is advisable to maintain docket communications up-to-date. This item is unresolved, pending NRC verification of a letter describing the deferral of these two commitments to the 1994 outage (50-244/93-03-01).

3.3 Conclusion

Based on the above, the design documentation and the safety analysis, the inspector concluded that the TCWS for the outage of the SW header "B" was adequate provided that conditions remain within the limits of the safety analysis, i.e., all the fuel is removed from the reactor vessel, and the SFP heat load does not exceed 14 MBTU/hr, and service water temperature does not exceed 60° F.

4.0 MANAGEMENT MEETINGS

Licensee management was informed of the scope and purpose of the inspection at the beginning of the inspection. The findings of the inspection were discussed with the licensee management at the February 19, 1993, exit meeting. See Attachment 1 for attendees.

ATTACHMENT 1

Persons Contacted

Rochester Gas and Electric Corporation

E. K. Voci	Manager Mechanical Engineering
* D. R. Markowski	Mechanical Engineer
* G. Wrobel	Manager Nuclear Safety
* B. J. Carrick	Lead Engineer
* R. Eliasz	Senior Nuclear Engineer
* R. Jaquin	Nuclear Safety Engineer
* C. Anderson	Manager of Quality Assurance
* S. T. Lawlor	Mechanical Engineer

U.S. Nuclear Regulatory Commission

T. Moslak	Senior Resident Inspector

*denotes those present at the exit meeting

FIGURE 1

EWR 5275 - CRFC Replacement Project

Comparison & Relationship of Heat Removal, Containment Pressure
and Fuel PCT Requirements & Design Limits

Fuel PCT Analyses		Containment Pressure Analyses				Containment Heat Removal (CRFCs)		
						Q (10^6 BTU/hr)		
		Pcont peak min ≥	Pback pass LOCA	Pcont peak min ≥	Pcont Peak calc new	UFSAR Limits	New CRFC Design	
Tpct Actual ≤	Tpct Current Design Analysis						150 (max)	M_{SW} = 1750 gpm M_{AIR} = 40 Kcfm T_{SW} = 35°F f = 0.000 BEST EST
						75 max CALC		
		Pcont peak min ≥	Pcont peak calc now					
							61.5 (Best Est)*	
			Pcont max <	Pcont * design			54.7 (Conservative Design)* Rating	
							51.9 (Purchase Spec Requirement)*	

*Pcont = 60 psig design

47 (min)

M_{SW} = 915 gpm
M_{AIR} = 33 Kcfm
T_{SW} = 80°F
f = 0.001