June 23, 1981

Docket No. 50-244
LS05-81-06-102

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JUN 26 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

RE: SEP TOPIC IX-3, GINNA

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topic IX-3, Station Service and Cooling Water Systems.

This assessment compares your facility, as described in Docket No. 50-244 with the criteria currently used by the regulatory staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessment within 30 days of receipt of this letter.

This evaluation will be a basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. This topic assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Draft SEP Topic IX-3

cc w/enclosure:
See next page

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
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Sincerely,

Richard [illegible] for

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosure:
Draft SEP Topic IX-3

cc w/enclosure:
See next page

DRAFT SEP REVIEW

OF

STATION SERVICE AND COOLING

WATER SYSTEMS

TOPIC IX-3

FOR THE

R. E. GINNA NUCLEAR POWER PLANT

I. INTRODUCTION

The safety objective of Topic IX-3 is to assure that the cooling water systems have the capability, with adequate margin, to meet the design objectives and, in particular, to assure that:

a. systems are provided with adequate physical separation such that there are no adverse interactions among those systems under any mode of operation;

b. sufficient cooling water inventory has been provided or that adequate provisions for makeup are available;

c. tank overflow cannot be released to the environment without monitoring and unless the level of radioactivity is within acceptable limits;

d. vital equipment necessary for achieving a controlled and safe shutdown is not flooded due to the failure of the main condenser circulating water system.

II. REVIEW CRITERIA

The current criteria and guidelines used to determine if the plant systems meet the topic safety objectives are those provided in Standard Review Plan (SRP) Sections 9.2.1, "Station Service Water System", and 9.2.2, "Reactor Auxiliary Cooling Water Systems."

III. RELATED SAFETY TOPICS AND INTERFACES

The scope of review for this topic was limited to avoid duplication of effort since some aspects of the review were performed under related topics. The related topics and the subject matter are identified below. Each of the related topic reports contains the acceptance criteria and review guidance for its subject matter.

II-2.A - Severe Weather Phenomena

II-3.B.1 - Flooding of Equipment

III-3.B - Flooding of Equipment (Failure of Underdrain System)

VI-7.D - Flooding of Equipment (Long Term Passive Failures)

III-3.C - Inservice Inspection of Water Control Structures

III-4.C - Internally Generated Missiles

III-5 - Mass and Energy Releases (High Energy Line Break)

VI-2.D - Mass and Energy Releases

III-6 - Seismic Qualification

III-12 - Environmental Qualification

VI-7.C.1 - Independence of Onsite Power

VII-3 - Systems Required for Safe Shutdown

VIII-2 - Diesel Generators

IX-1 - Fuel Storage

IX-6 - Fire Protection

The following topics are dependent on the present topic information for completion:

VI-3 - Containment Pressure and Heat Removal Capability

IX-5 - Ventilation Systems

XV-7 - Reactor Coolant Pump Rotor Seizure

IV. REVIEW GUIDELINES

In addition to the guidelines of SRP Sections 9.2.1 and 9.2.2, in determining which systems to evaluate under this topic the staff used the definition of "systems important to safety" provided in Reference 1. The definition states systems important to safety are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary*, (2) the capability to shutdown the reactor and maintain it in a safe condition, or (3) the capability to prevent, or mitigate the consequences of, accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100, "Reactor Site Criteria." Since we consider essential systems as systems or portions of systems important to safety we used the definition to identify the essential systems. It should be noted that this topic will be updated if future SEP reviews identify additional cooling water systems that are important to safety.

V. EVALUATION

In the course of this topic review, the staff considered the need to evaluate the Air Conditioning Chilled Water System and the Reactor Makeup Water System. The Air Conditioning Chilled Water System, which supplies water to ventilation coolers in the Control Room and Service Building, may be an essential system (for Control Room cooling); however, it will not be evaluated under this topic until the completion of SEP Topic IX-5, "Ventilation Systems" for Ginna at which time the safety significance of the system will be ascertained. The Reactor Water Makeup System was

*Reactor Coolant Pressure Boundary is defined in 10 CFR Part 50 §50.2(v).

considered for its function of supplying water to the reactor coolant pump (RCP) standpipes. Reactor makeup water is supplied to the RCP standpipes on an intermittent basis, when the standpipe loses volume due to evaporation. Its purpose is to supply pressure downstream of the second stage RCP seal and thus force some minor amount of water up through the third stage RCP seal. This function is not considered important to safety.

The systems which were reviewed under this topic are the Component Cooling Water System and the Service Water System. The Spent Fuel Pool Cooling System is discussed in the SEP review of Topic IX-1, "Fuel Storage."

V.I <u>Component Cooling Water System</u>

The Component Cooling Water (CCW) system removes heat from various plant systems and components and transfers this heat to the Service Water System. The heat loads on the system are:

1. Residual Heat Removal (RHR) heat exchangers

2. Emergency Core Cooling System (ECCS) pumps

    a. Residual Heat Removal pumps
    b. High Pressure Safety Injection pumps
    c. Containment Spray pumps

3. Reactor Coolant Pumps

4. Non-regenerative heat exchanger

5. Excess letdown heat exchanger

6. Reactor support cooling pads

7. Seal Water heat exchanger

8. Sample heat exchanger

9. Boric Acid evaporator

10. Waste gas compressors

11. Waste evaporator

During normal plant operation, one CCW pump and one CCW heat exchanger are in operation, and they can accommodate the heat removal load on the system. Both pumps and heat exchangers are normally used for a plant cooldown; however, if one pump or one heat exchanger is not operable, safe operation of the plant is not affected, but the time to cool the plant is extended (Reference 2). CCW pump A and B receive electrical power from 480 V buses 14 and 16, respectively.

The staff reviewed the heat removal requirements of the CCW system during post-accident conditions. The accidents considered were the Loss of Coolant Accident (LOCA) and the Main Steam Line Break(MSLB) Inside Containment because these events result in the greatest potential accident heat loads on the CCW system. The Containment Fan Coolers are also discussed here because they complement the CCW system in the post-accident containment heat removal function. Section 14.3.4 of Reference 3 provides an analysis of containment integrity following a LOCA. Some part of the energy available for release to the containment must be removed to prevent exceeding the containment design pressure limit.* Energy is removed by the fan coolers and the Containment Spray (CS) system. The fan coolers transfer heat from the containment atmosphere to the Service Water System. The CS system removes heat from the containment by spraying cool water directly into the containment atmosphere. This water, now heated, drains to the containment sump. The heat is then transferred to the CCW system through the Residual Heat Removal (RHR) heat exchangers when the containment sump fluid is pumped, by the RHR system, back to the CS system during the post-accident recirculation mode of ECCS operation. Two fan coolers and one CS pump are supplied power from separate 480 V emergency buses. The minimum combination of containment cooling systems occurs after a postulated loss of offsite power and the failure of one of the two emergency diesel generators. This minimum combination (1 CS pump and 2 fan coolers) was the combination analyzed in Reference 3. Using the design parameters of the CS, CCW, and SWS, shown in Table 1, the containment analysis of Reference 3 concluded that the heat load which must be removed from containment can be accommodated by the CS and fan coolers given the assumed failure of either diesel generator.

During the initial period of energy release to the containment, the heat is absorbed in various passive heat sinks inside containment. When the containment heat removal systems are initiated, the SWS is the first cooling water system to assume a heat load since the fan coolers start to remove heat directly from the containment atmosphere. At the design heat removal rate of the fan coolers (Table 1), the SWS temperature at the cooler exit is 174°F which is well below the design temperature limit of 200°F. The heat removed from the containment by the CS system is collected in the containment sump until the recirculation phase of the accident commences (after from 31 to 54 minutes). During recirculation, the sump fluid is recycled via the RHR system (and the RHR heat exchangers) back to the CS pumps for reuse. The sump water temperature would be approximately 250°F at the start of recirculation (Reference 3, Appendix 6E, Figure 1). Using Figure 14.3.4-9 of Reference 3 to determine the rate of energy addition to the containment at start of recirculation (119 E6 BTU/hr), the staff performed a heat balance on the CS, CCW

*The SEP will reevaluate the post-accident energy balance in containment under Topic VI-2.D, "Mass and Energy Release for Postulated Pipe Breaks Inside Containment."

and SWS (Figure 1). The results of this calculation show that the CCW and SWS have sufficient capability to remove the post-LOCA heat loads with approximately 47 E6 BTU/HR accommodated by CS and approximately 72 E6 BTU/hr by the fan coolers at the start of recirculation. The normal system lineup is shown in Figure 1 for initial recirculation operation with a failed diesel generator. The heat loads are accommodated without exceeding RHR or CCW design temperatures which are 400°F and 200°F, respectively, and without operator action to adjust SWS or CCW system flows.

For the MSLB inside containment event, the amount of energy added to the containment should be no greater than that added for the LOCA case. (Ongoing SEP reviews will verify that the assumptions used to determine the magnitude of energy addition to the containment are acceptable.) Because the safety injection flow to the reactor coolant system would not be available as a heat sink inside containment following a MSLB, the containment sump would be filled by condensed superheated steam from the MSLB and CS water; and a higher sump fluid temperature would be achieved earlier in the MSLB case than in the post-LOCA case. This would not affect the heat load on the CCW system however; because, if recirculation of the containment sump fluid were necessary, it would not be initiated by the operator until much later into the MSLB accident sequence when containment sump level would be approximately equivalent to the level when recirculation would be initiated following a LOCA. Given the smaller heat release to containment following a MSLB and approximately equal sump levels at the start of recirculation following both the MSLB and LOCA, the heat load on the CCW system is expected to be no greater than the heat load following a LOCA.

In the post-accident case, the potentially most limiting single failure from the standpoint of CCW high temperature would be the failure in the open position of the motor operated CCW supply valve to an idle RHR heat exchanger. With half of the CCW flow diverted to an idle heat exchanger, the temperature of the CCW at the exit of the active heat exchanger could exceed 200°F. This condition can be remedied by the control room operator by increasing RHR heat exchanger bypass flow and thus reducing the energy removal rate of the active RHR heat exchanger. The fan coolers have the capacity and would pick up the additional heat load from containment.

No post-accident realignment of the CCW system is performed by the operator except for the opening of a CCW supply valve to one RHR heat exchanger at the start of recirculation and closing, or verifying the automatic closure of, the isolation valves to all and each of the services inside containment. These actions can be performed from the control room.

The RHR, CS, CCW and SWS pumps and valves are powered from the appropriate emergency buses such that a failure of one bus would not prevent the operation of the systems as analyzed in the post-accident condition.

During normal CCW system operation, single active failure could prevent flow to all or each of the services inside containment (Reactor Coolant Pumps, Excess Letdown Heat Exchanger, and Reactor Support Cooling). Loss of flow to the containment services requires prompt operator action to prevent damage to the Reactor Coolant Pumps (RCP). Damage to an RCP from loss of cooling flow could result in pump seizure and cause a loss of flow accident. (The consequences of a postulated RCP seizure are evaluated as an SEP Design Basis Event.) Plant procedures require the operator to trip the reactor and then the RCP's within two minutes following a loss of CCW flow to the pumps or before RCP motor bearing temperature reaches 200°F. Plant shutdown following reactor trip is in accordance with established emergency procedures. Loss of CCW flow to the excess letdown heat exchanger or reactor supports does not require immediate operator action, but, if CCW flow cannot be restored to these components, the reactor plant must eventually be shut down.

Isolation of individual leaking components is accomplished, with the exception of components inside containment, by manual valves. Also, although the CCW pumps and heat exchangers are redundant, they are connected by single pipe headers whose failure could disable the system. However, at the operating pressure and temperature of the system (100 psig, 200°F), a passive failure would most probably result in a leak rate which the staff estimates to be 210 gpm using the methods of Reference 5 for a 10" pipe. The normal volume of water in the surge tank (1000 gal.) would provide the operators with about 5 minutes at a leak rate of 210 gpm to stop a leak from the system. It is improbable that the operator could act within this time period, and it is possible that the leak may be in an unisolable portion of the system. If a loss of the CCW system occurs during normal plant operation, the licensee has an operating procedure that directs the operator to shutdown the reactor and commence decay heat removal using the steam generators with natural circulation of the reactor coolant system. If CCW cannot be readily restored, a plant cooldown would be commenced. For a cooldown with no CCW, the cooldown method and system described in Reference 6 (with the exception of the CCW and RHR systems) would be available, and the licensee has proposed a method to achieve cold shutdown conditions independent of the CCW and RHR systems using the steam generators (Ref. 9).

Loss of the CCW system during post-accident operation was considered in the Provisional Operating License review of Ginna, and the staff concluded that the RHR pumps could continue to operate to recirculate containment sump water with decay heat being removed by the containment fan coolers. However, because the CCW system cools the bearings and lubricating oil coolers for the RHR (and other ECCS) pumps, these pumps would not be available to recirculate the sump water. Current criteria for piping system passive failures do not require the assumed passive failures of moderate energy systems (like the CCW) under post-accident conditions, although system leaks are assumed (Ref. 7). Therefore, the CCW system makeup capability must only be required to cope with normal system leakage in post-accident operation.

We also considered the effects of such a loss of CCW during a cooldown of the plant with the RHR system operating. In this case, with the reactor vessel head installed, the RCS temperature would rise to greater than 200^{O}F and decay heat could continue to be removed via the steam generator atmosphere relief valves using natural circulation. Steam generator feed would be accomplished by the Auxiliary Feed System (AFS). The plant could remain in this condition while CCW repairs were made. For normal decay heat removal when the reactor vessel head is removed, adequate cooling can be provided by keeping the core flooded (using various systems such as RHR and CVCS) while repairs are made to the CCW piping. The CCW system is accessible for repairs and can be filled with water in less than two hours after the repairs are completed starting with a completely drained system (Reference 3, page 9.3-18).

During normal and post-accident operation, thermal expansion and contraction of the CCW system liquid is accommodated by the CCW surge tank, and leakage into or out of the system can be detected by surge tank level changes. High and low surge tank levels are alarmed in the control room, and a radiation monitor and alarm alerts the control room operator to the leakage of radioactive fluid into the CCW system from components which contain reactor coolant. The surge tank also maintains a positive suction head on the CCW pumps during normal and post accident operation. Makeup water to the CCW system is supplied by either the demineralized water system or the reactor makeup water system via local manual valves in the auxiliary building. The makeup rate is sufficient to accommodate system leakage but not to accommodate a moderate energy line break per Reference 5; however, the seismic classification of the CCW makeup supply systems is not sufficient to assure the availability of makeup water following a seismic event. The licensee may be required to provide assurance that CCW makeup can be supplied following an earthquake.

Based on our review of the CCW system, the safety related functions are to provide cooling for the RHR heat exchangers, ECCS pumps, RCPs, and reactor support cooling pads. Of these functions only RHR heat exchanger and ECCS pump cooling are considered to be essential functions. Plant procedures provide adequate protection from the effects of losing the other safety related functions.

V.II <u>Service Water System</u>

The Service Water System (SWS) circulates water from the screen house on Lake Ontario to various heat exchangers and systems in the containment, auxiliary and turbine buildings. These buildings are Class I structures except for the turbine building. The system has four pumps, three of which are in operation during normal operating conditions. As described in the previous CCW section, two SWS pumps are required to remove heat from components under post-accident conditions.

The SWS piping is arranged so that there are two flow paths to the redundant "critical"* loads identified in Table 2. Another header supplies various "non-critical" loads (see Table 3). The "non-critical" loads are automatically isolated from the "critical" headers by redundant motor operated valves when a reactor safeguards actuation signal occurs. Redundant motor operated isolation valves also automatically secure SWS flow to the air conditioning chill water system, circulating water pumps, and screen wash supply on a safeguards actuation signal.

During normal plant operation, the SWS supplies flow to all loads except the auxiliary standby auxiliary feed systems. During RHR operation for a normal plant cooldown, almost all "non-critical" loads may be removed from the SWS, if necessary. Following a safeguards actuation signal, the SWS supplies all "critical" loads except the backup feedwater supply to the auxiliary and standby auxiliary feed systems, which require operator action to receive SWS flow.

To overcome single failures in the system each "critical" load has a redundant counterpart cooled by the other "critical" SWS header. If necessary, an operator could cross-connect the "critical" headers by means of manual valves to achieve added system flexibility. In the normal system alignment, no single active or passive failure could result in the loss of SWS flow to redundant "critical" loads except for the reactor vessel cavity coolers which could both be disabled by a single passive failure. Since the SWS is a moderate energy system, a passive pipe failure would probably result in a leak rather than a complete pipe rupture. Using the method described in Reference 5, the estimated leakage from a SWS header is 585 gpm for a 20" header at 75 psig. Although this leak may pose a flooding problem, the supply function of the affected header would not be significantly impaired.** A leak from the 2.5" supply line to the reactor cavity coolers would result in the loss of about 25 gpm. This leak rate would not completely disable the coolers which normally receive about 45 gpm of SWS flow.

Leak detection for the SWS is provided by header pressure switches and by sump level alarms in the buildings which house the SWS (see the SEP review of Topic III-5.B). Isolation of leaking components is accomplished by closing manual valves, in general; however, remotely operated valves can secure SWS flow to "non-critical" loads, each CCW heat exchanger, and the spent fuel pool cooling system.

Electrical power for the SWS pumps is provided by 480 V buses 17 and 18. One SWS pump is started on each diesel generator during post-accident generator load sequencing.

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*"Critical" refers to a heat load that the licensee has designated as safety related.

**The effects of flooding from pipe leaks will be reviewed under SEP Topic III-5.B, "Pipe Breaks Outside Containment."

A review of Licensee Event Reports shows no recurring problems with operation or maintenance of the SWS (or CCW system).

Based on our review of the SWS, we consider the components supplied by the "critical" supply headers (Table 2) to be the essential loads on the system.

VII. CONCLUSION

Based on our review of the service and cooling water systems for Ginna, we have concluded that the essential systems and functions are:

Component Cooling Water: RHR heat exchanger cooling and ECCS pump cooling.

Service Water System: All components supplied by the "critical" supply headers (Table 2).

We have determined that the design of the above systems is in conformance with current regulatory guidelines and with General Design Criterion (GDC) 44 regarding capability and redundancy of the essential functions of the systems. The systems also meet the requirements of GDC 45 and 46 regarding system design to permit periodic inspection and testing.

With respect to the seismic design of the CCW makeup supply systems the licensee may be required to provide assurance that CCW makeup can be supplied following an earthquake. This will be determined in the integrated safety assessment for the facility.

As was indicated in Section III of this evaluation, the spent fuel pool cooling system is being reviewed in SEP Topic IX-1. That review will also address the proposed modifications to the spent fuel pool cooling system. If the findings of Topic IX-1 necessitate any additional review of the SWS, it will be addressed in the integrated safety assessment for the facility.

TABLE 1. SYSTEM DESIGN PARAMETERS

System/Reference	Parameters
Containment Spray (Ref. 3, Section 6.4)	2 pumps - 1250 gpm each 2 RHR heat exchangers - 24.15E6 BTU/hr each (with 1525 gpm RHR @ 160°F and 2780 gpm CCW @ 100°F)
Component Cooling (Ref. 3, Section 9.3)	2 pumps - 2980 gpm each 2 CCW heat exchangers - 24.15E6 BTU/hr each (with 2950 gpm CCW @ 117° and 5055 gpm SWS @ 80°F)
Service Water (Ref. 3, Section 9.6)	4 pumps - 5300 gpm each
Containment Fan Coolers (Ref. 3, Section 5.3, 6.3, and 9.6)	4 Units - 13,900 BTU/sec each* (at 60 psig, 286°F and 4248 gpm SWS)

*Evaluated in Reference 4.

TABLE 2. "CRITICAL" SWS LOADS

1. Safety injection pump bearings
2. ECCS and charging pump ventilation coolers
3. Component Cooling Water heat exchangers
4. Containment Fan Coolers (and motors)
5. Diesel generators
6. Auxiliary Feed System (feedwater supply and pump cooling)
7. Standby Auxiliary Feed System
8. Spent Fuel Pool Cooling
9. Reactor vessel cavity coolers
10. Containment penetration cooling

TABLE 3. "NON-CRITICAL SWS LOADS

1. Plant air compressors
2. Condensate and heater drain pumps
3. Relay room air conditioning
4. Turbine exciter
5. Bus duct coolers
6. Seal oil unit
7. Blowdown tank
8. Sample coolers
9. Electro-hydraulic control system
10. Turbine oil coolers
11. Feed pump oil coolers
12. Containment pressure testing air cooler
13. Vacuum pumps

14. Fire booster pump supply

15. Circulating Water pumps

16. Screen wash supply

17. Air conditioning chill water coolers

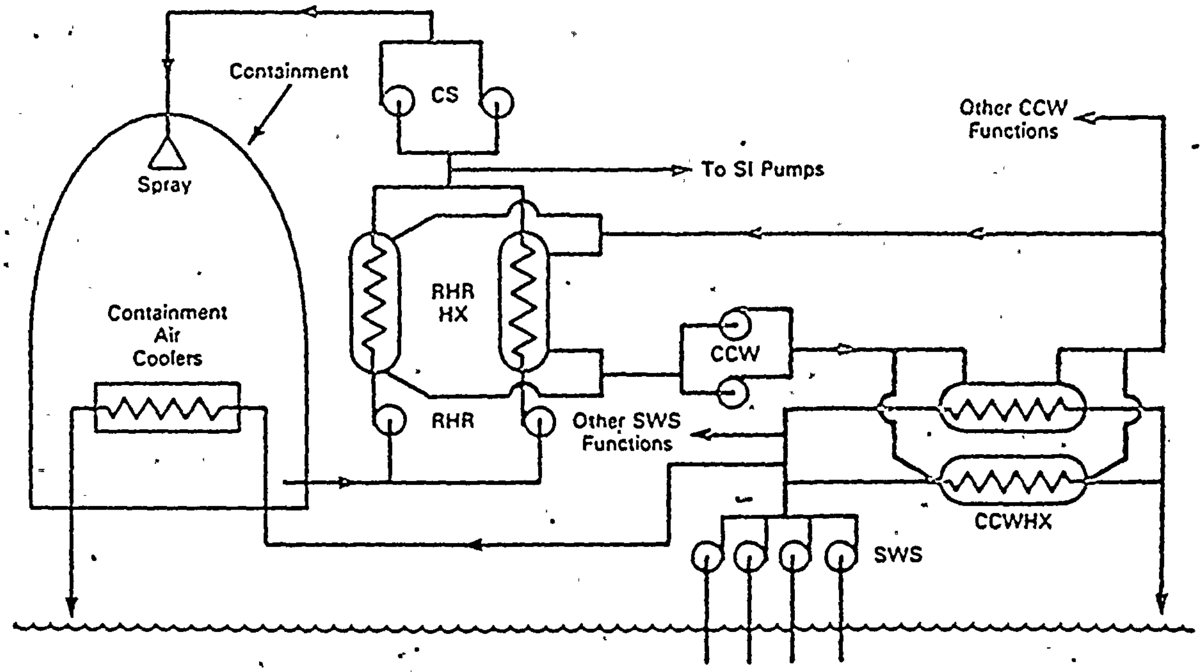
Containment
CS
Other CCW
Functions
Spray
To SI Pumps
RHR
HX
Containment
Air
Coolers
CCW
RHR
Other SWS
Functions
CCWHX
SWS

FIGURE 1

3.0 REFERENCES

1. Regulatory Guide 1.105, "Instrument Setpoints."

2. Fire Protection Evaluation, Robert E. Ginna Nuclear Power Plant, Unit 1, transmitted by RG&E letter dated March 24, 1977 (Section 3.2.6).

3. Robert Emmett Ginna Nuclear Power Plant, Unit No. 1, Final Facility Description and Safety Analysis Report.

4. Addendum to the Safety Evaluation by the Division of Reactor Licensing, USAEC, for the R. E. Ginna Nuclear Power Plant, dated September 19, 1969.

5. Branch Technical Position MEB 3-1, appended to Standard Review Plan 3.6.2.

6. SEP Review of Safe Shutdown Systems for the R. E. Ginna Nuclear Power Plant (SEP Topics VII-3, V-10.B, V-11.A, V-11.B, X).

7. Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR to NRR Staff, NUREG-0153, Issue #17, December 1976.

8. SEP Review of Residual Heat Removal System Heat Exchanger Tube Failures, Topic V-10.A.

9. RG&E letter L. White to D. Ziemann, dated December 28, 1979, transmitting Fire Protection-Shutdown Analysis, R. E. Ginna Nuclear Power Plant.