ATTACHMENT A

Replace Technical Specification pages 3.3-2, 3.3-4, 3.3-7, 3.15-1, 3.15-2, 4.4-8, 4.6-3, 4.6-4, and 4.8-2.

a. The refueling water tank contains not less than 300,000 gallons of water, with a boron concentration of at least 2,000 ppm.

b. Each accumulator is pressurized to at least 700 psig with an indicator level of at least 50% and a maximum of 82% with a boron concentration of at least 1800 ppm. Neither accumulator may be isolated.

c. Three safety injection pumps are operable.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All valves, interlocks and piping associated with the above components which are required to function during accident conditions are operable.

g. A.C. Power shall be removed from the following valves with the valves in the open position: safety injection cold leg injection valves 878B and D, accumulator injection valves 841 and 865, and refueling water storage tank delivery valves 856. A.C. power shall be removed from safety injection hot leg injection valves 878A and C with the valves closed. D.C. control power shall be removed from refueling water storage tank delivery valves 896A and B with the valves open. In the meantime, single failure protection for valves 896A and B will be provided by locking out A.C. power, remote from the control room, with operating personnel assigned specifically to restore A.C. power when the valves are required to function in the event of a loss-of-coolant accident.

h. Revisions to procedures for post-LOCA long term cooling as described in letters to the Nuclear Regulatory Commission from Rochester Gas and Electric Corporation datd April 1, 1975, and May 13, 1975, shall be implemented prior to reactor startup following the shutdown of March 10, 1975.

i. Check valves 853A, 853B, 867A, 867B, 878G, and 878J shall be operable with less than 5.0 gpm leakage each. The leakage requirements of Technical Specification 3.1.5.1 are still applicable.

d. One residual heat exchanger may be out of service for a period of no more than 72 hours.

e. Any valve, interlock, or piping required for the functioning of one safety injection train and/or one low head safety injection train (RHR) may be inoperable provided repairs are completed within 72 hours. Prior to initiating valve repairs, all valves in the system that provide the duplicate function shall be tested to demonstrate operability.

f. Power may be restored to any valve referenced in 3.3.1.1 g for the purposes of valve testing providing no more than one such valve has power restored and provided testing is completed and power removed within 12 hours.

g. Those check valves specified in 3.3.1.1 i may be inoperable (greater than 5.0 gpm leakage) provided the inline MOVs are de-energized closed and repairs are completed within 12 hours.

3.3.1.3 Except during diesel generator load and safeguard sequence testing or when the vessel head is removed or the steam generator manway is open, no more than one safety injection pump shall be operable whenever the overpressure protection system is required to be operable.

3.3.1.3.1 Whenever only one safety injection pump may be operable by 3.3.1.3, at least two of the three safety injection pumps shall be demonstrated inoperable a minimum of once per twelve hours by verifying that the control switches are in the pull-stop position.

3.3.2 <u>Containment Cooling and Iodine Removal</u>

3.3.2.1 The reactor shall not be made critical except for low temperature physics tests, unless the following conditions are met:

a. The spray additive tank contains not less than 4500 gallons of solution with a sodium hydroxide concentration of not less than 30% by weight.

b. At least two containment spray pumps are operable.

c. Four fan cooler units are operable.

3.15 Overpressure Protection System

Applicability

Applies whenever the temperature of one or more of the RCS cold legs is $\leq$ 330°F, or the Residual Heat Removal System is in operation.

Objective

To prevent overpressurization of the reactor coolant system and the residual heat removal system.

Specification

3.15.1 Except during secondary side hydrostatic tests in which RCS pressure is to be raised above the PORV setpoint, at least one of the following overpressure protection systems shall be operable:

a. Two pressurizer power operated relief valves (PORVs) with a lift setting of $\leq$ 435 psig, or

b. A reactor coolant system vent of $\geq$ 1.1 square inches.

3.15.1.1 With one PORV inoperable, either restore the inoperable PORV to operable status within 7 days or depressurize and vent the RCS through a 1.1 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.2 With both PORVs inoperable, depressurize and vent the RCS through a 1.1 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to operable status.

3.15.1.3 Use of the overpressure protection system to mitigate an RCS or RHRS pressure transient shall be reported in accordance with 6.9.3.

Basis

The operability of two pressurizer PORVs or an RCS vent opening of greater than 1.1 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold

in the hot shutdown condition. If the requirements of 3.3.3.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition.

a. One component cooling pump may be out of service provided the pump is restored to operable status within 24 hours.

b. One heat exchanger or other passive component may be out of service provided the system may still operable at 100% capacity and repairs are completed within 24 hours.

3.3.4 Service Water System

3.3.4.1 The reactor shall not be made critical unless the following conditions are met:

a. At least two service water pumps, one on bus 17 and one on bus 18, and one loop header are operable.

b. All valves, interlocks, and piping associated with the operation of two pumps are operable.

3.3.4.2 Any time that the conditions of 3.3.4.1 above cannot be met, the reactor shall be placed in the cold shutdown condition.

3.3.5 Control Room Emergency Air Treatment System

3.3.5.1 The reactor shall not be made critical unless the control room emergency air treatment system is operable.

legs are $\leq 330^{\circ}F$. This relief capacity will also ensure that no overpressurization of the RHR system could occur. Either PORV has adequate relieving capability to protect the RCS and RHRS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator $\leq 50^{\circ}F$ above the RCS cold leg temperature or (2) the start of a safety injection pump and its injection into a water solid RCS.[1,2]

References:

(1) L. D. White, Jr. letter to A. Schwencer, NRC, dated July 29, 1977

(2) SER for SEP Topics V-10.B, V-11.B, VII-3, "Safe Shutdown," dated September 29, 1981

the tendon containing 6 broken wires) shall be inspected. The acceptance criterion then shall be no more than 4 broken wires in any of the additional 4 tendons. If this criterion is not satisfied, all of the tendons shall be inspected and if more than 5% of the total wires are broken, the reactor shall be shut down and depressurized.

4.4.4.2 Pre-Stress Confirmation Test

a. Lift-off tests shall be performed on the 14 tendons identified in 4.4.4.1a above, at the intervals specified in 4.4.4.1b. If the average stress in the 14 tendons checked is less than 144,000 psi (60% of ultimate stress), all tendons shall be checked for stress and retensioned, if necessary, to a stress of 144,000 psi.

b. Before reseating a tendon, additional stress (6%) shall be imposed to verify the ability of the tendon to sustain the added stress applied during accident conditions.

4.4.5 Containment Isolation Valves

4.4.5.1 Each isolation valve specified in Table 3.6-1 shall be demonstrated to be operable in accordance with the Ginna Station Pump and Valve Test Program submitted in accordance with 10 CFR 50.55a.

4.4.6 Containment Isolation Response

4.4.6.1 Each containment isolation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.4.6.2 The RESPONSE TIME of the containment isolation valves, as listed in Table 3.6-1, shall be demonstrated to be within the limit at least once per 18 months. This response time includes only the valve travel times for all valves that change position.

Basis:

The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions, the temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286°F.

c. As each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

d. Each battery shall be subjected to a load test within a twelve-month period from the last load test; however, to permit the load test to coincide with a scheduled refueling, the period may extend for an additional three months. The battery voltage as a function of time shall be monitored to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

e. Each battery shall be subject to a discharge test at least once each 60 months. The purpose of this test is to show that the battery capacity is at least 80% of the manufacturer's recommendations.

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safeguards equipment will function automatically in the event of a loss of all normal 480V AC station service power.(1)

The testing frequency specified will be often enough to identify and correct any mechanical or electrical deficiency before it can result in a system failure. The fuel supply and starting circuits

and controls are continuously monitored and any faults are indicated by alarm. An abnormal condition in these systems would be signaled without having to place the diesel generators themselves on test.

Station batteries will deteriorate with time, but precipitous failure is extremely unlikely. The surveillance specified is that which has been demonstrated over the years to provide an indication of a cell becoming unserviceable long before it fails, and to ensure that the battery capacity is acceptable.

The equalizing charge, as recommended by the manufacturer, is vital to maintaining the ampere-hour capability of the battery. As a check upon the effectiveness of the equalizing charge, the battery should be loaded rather heavily and the voltage monitored as a function of time. If a cell has deteriorated or if a connection is loose, the voltage under load will drop excessively indicating replacement or maintenance.

The minimum permissible on-site fuel oil inventory, 10,000 gallons, is sufficient for operation under loss-of-coolant accident conditions of two engineered safety features trains for 48 hours, or for one train for 80 hours, or for operation under hot standby non-accident conditions for 111 hours.[2]

<u>References</u>:

(1) FSAR, Section 8.2

(2) FSAR, Section 8.2.3

4.8.5 Except during cold or refueling shutdowns, the suction, discharge, and cross-over motor operated valves for the Standby Auxiliary Feedwater pumps shall be exercised at intervals not to exceed one month.

4.8.6 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly. These tests shall be performed prior to exceeding 5% power during a startup if the time since the last test exceeds one month.

4.8.7 At least once per 18 months, control of the standby auxiliary feed system pumps and valves from the control room will be demonstrated.

4.8.8 At least once per 18 months during shutdown

a. Verify that each automatic valve in the flow path for each auxiliary feedwater pump actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

b. Verify that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

4.8.9 Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.1-1.

4.8.10 The RESPONSE TIME of each pump and valve required for the operation of each "train" of auxiliary feedwater shall be demonstrated to be within the limit of 10 minutes at least once per 18 months.

Basis

The monthly testing of the auxiliary feedwater pumps by supplying feedwater to the steam generators will verify their ability to meet design. The flow rates will be measured at a simulated steam generator pressure of 1100 psia. The capacity of any one of the three auxiliary feedwater pumps is sufficient to meet decay heat removal requirements. Proper functioning of the steam turbine admission valve and the feedwater pumps start will demonstrate the integrity of the steam driven pump.

Monthly testing of the Standby Auxiliary Feedwater pumps by supplying water from a condensate supply tank to the steam generators will verify their ability to meet design. The flow rate will be measured at a simulated steam generator pressure of 1100 psia.

ATTACHMENT B
Safety Evaluation

1. The proposed Technical Specification change on pages 3.15-1 and 3.3-4 regarding the operability of the Overpressure Protection System (OPS) will ensure that the Residual Heat Removal System will also be protected from any postulated pressure excursions. The change will provide for the Overpressure Protection System to be made operable whenever the Residual Heat Removal System is placed in operation. The present procedures specify that the RHR system can be placed in operation at 360 psig and 350°F, while the OPS need only be made operable when the RCS cold leg temperature is less than or equal to 330°F.

   The Technical Specification change was originated as a result of the NRC review of SEP Topic V-10.B, "RHR Reliability," and is also specified in NUREG-0821, Section 4.21.1.

   Since this change provides protection for the RHR system, and maintains protection of the Reactor Coolant System, this Technical Specification change will not result in any undue risk to the health and safety of the public.

2. The proposed Technical Specification change on p. 3.3-2 modifies the minimum quantity of borated water required in the refueling water storage tank from 230,000 gallons to 300,000 gallons. As determined in SEP Topic VI-7.B, "ESF Switchover from Injection to Recirculation Mode," and stated in NUREG-0821, Section 4.23.1, this quantity of water will provide the necessary NPSH for the RHR pumps under post-LOCA recirculation conditions.

   Since this change provides an increased minimum water level for the RHR pumps in containment, this Technical Specification change will not result in any undue risk to the health and safety of the public.

3. The proposed Technical Specification change on pages 4.4-8 and 4.8-2 will delete the requirements for response time testing of the initiating circuitry in paragraphs 4.4.6.2 and 4.8.10 for containment isolation valves and the auxiliary feedwater system, respectively. RG&E has conducted this testing, from the sensor through the bistable devices, during the 1981 and 1982 refueling outages, and found that the testing does not appear to be beneficial. This particular portion of the overall system response time is a very small fraction of the total system response time (milliseconds vs. 1 to 10 minutes). However, valuable personnel time is expended to perform this testing. RG&E proposes that functional testing of the actuated equipment be retained, and response time testing of critical items, such as containment

isolation valve stroke times, diesel generator start and sequencing times, pump start times, and rod drop time be performed. The sensor to actuated equipment bistable testing would be deleted, however. This RG&E proposal is comparable to the finding made for other plants during the Integrated Assessment. For example, in NUREG-0820, it was concluded that, from a risk perspective, the effect of including this additional testing is negligible. Backfitting was not recommended.

Since this change merely deletes testing which is not considered to be beneficial, this Technical Specification change will not result in any risk to the health and safety of the public.

4. The proposed Technical Specification change on page 3.3-7 will clarify the electrical power alignment of the service water pumps. During the review of SEP Topic IX-3, "Station Service and Cooling Water Systems," it was noted that, during power operation, the Ginna Technical Specifications require that two service water pumps be operable. It does not specify that at least one of these pumps be aligned to each of the two redundant Class IE power supplies. This Technical Specification will clarify this issue. Since this change will merely clarify the Class IE power alignment of these pumps, RG&E concludes that this Technical Specification change will not result in any risk to the health and safety of the public.

5. The proposed Technical Specification change on page 4.6-3 will add a battery discharge test requirement, to be performed at least once each 60 months, to verify that the battery capacity is at least 80% of the manufacturer's rating. The purpose of the discharge test is to provide information about the battery capacity, and to disclose the possible degradation of any battery cells. This issue was discussed in SEP Topic VIII-3.A, "Station Battery Capacity Test Requirements," and is noted in NUREG-0821, Section 3.3.8. RG&E performed this discharge test during the 1982 refueling outage.

6. Technical Specifications to address the definition of "operability" and other changes required to implement this change will be submitted separately.

ATTACHMENT C

In promulgating its interim final rule to comply with Public Law 97-415, the Commission published as guidelines a series of examples of amendments that are not likely to involve significant hazards considerations. 48 Fed. Reg. 14864, 14870. The Commission's example (ii) relates to changes that constitute an additional limitation, restriction or control not presently included in the technical specifications. Four of the five changes in Technical Specifications proposed in the amendment clearly come within example (ii). The fifth change, deletion of the process-to-actuator response from testing requirement for auxiliary feedwater and containment isolation, comes within the intent of example (vi), a change that may reduce in some way a safety margin, but the results of the change are clearly within all acceptable criteria. [Note: example (iv) may also apply.] Since this portion of the overall system response involves a very small fraction of the total system response time and since the NRC Staff has found in other situations, that from a risk perspective the effect of the additional testing is negligible, the change is not a significant relaxation in the Technical Specifications.

None of the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety.