ATTACHMENT A

Revise the Technical Specification pages as follows:

Remove	Insert
3.6-2	3.6-2
3.6-3	3.6-3
3.6-4	-----
3.6-5	-----
3.6-6	-----
3.6-7	-----
3.6-7A	-----
3.6-8	-----
3.6-9	-----
3.6-10	-----
3.6-11	-----
4.4-4	4.4-4
4.4-7	4.4-7
4.4-8	4.4-8
4.4-11	4.4-11

3.6.3 Containment Isolation Valves

3.6.3.1 With one or more containment isolation valves inoperable, maintain at least one isolation boundary OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to operable status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Containment isolation valves are inoperable from a leakage standpoint when the demonstrated leakage of a single containment isolation valve or cumulative total leakage of all containment isolation valves is greater than that allowed by 10 CFR 50 Appendix J.

3.6.4 Combustible Gas Control

3.6.4.1 When the reactor is critical, at least two independent containment hydrogen monitors shall be operable. One of the monitors may be the Post Accident Sampling System.

3.6.4.2 With only one hydrogen monitor operable, restore a second monitor to operable status within 30 days or be in at least hot shutdown within the next 6 hours.

3.6.4.3 With no hydrogen monitors operable, restore at least one monitor to operable status within 72 hours or be in at least hot shutdown within the next 6 hours.

3.6.5 Containment Mini-Purge

Whenever the containment integrity is required, emphasis will be placed on limiting all purging and venting times to as low as achievable. The mini-purge isolation valves will remain closed to the maximum extent practicable but may be open for pressure control, for ALARA, for respirable air quality considerations for personnel entry, for surveillance tests that may require the valve to be open or other safety related reasons.

Basis:

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there would be no pressure buildup in the containment if the reactor coolant system ruptures.

The shutdown margins are selected based on the type of activities that are being carried out. The (2000 ppm) boron concentration provides shutdown margin which precludes criticality under any circumstances. When the reactor head is not to be removed, a cold shutdown margin of 1%Δk/k precludes criticality in any occurrence.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major steam break accident were as much as 1 psig.[1] The containment is designed to withstand an internal vacuum of 2.5 psig.[2] The 2.0 psig vacuum is specified as an operating limit to avoid any difficulties with motor cooling.

Containment isolation valves and boundaries are listed in UFSAR Table 6.2-13. Isolation boundaries can include isolation valves, closed systems, and flanges as shown on UFSAR Table 6.2-13.

References:

(1) Westinghouse Analysis, "Report for the BAST Concentration Reduction", August 1985

(2) UFSAR - Section 3.8.1.2.2

(3) UFSAR Table 6.2-13

4.4.1.4 Acceptance Criteria

a. The leakage rate Ltm shall be <0.75 Lt at Pt. Pt is defined as the containment vessel reduced test pressure which is greater than or equal to 35 psig. Ltm is defined as the total measured containment leakage rate at pressure Pt. Lt is defined as the maximum allowable leakage rate at pressure Pt.

b. Lt shall be determined as $Lt=La\left(\frac{Pt}{Pa}\right)^{1/2}$ which equals .1528 percent weight per day at 35 psig. Pa is defined as the calculated peak containment internal pressure related to design basis accidents which is greater than or equal to 60 psig. La is defined as the maximum allowable leakage rate at Pa which equals .2 percent weight per day.

c. The leakage rate at Pa (Lam) shall be <0.75 La. Lam is defined as the total measured containment leakage rate at pressure Pa.

4.4.1.5 Test Frequency

a. A set of three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted in the final year of the 10-year service period or one year before or after the final year of the 10-year service period provided:

   i. the interval between any two Type A tests does not exceed four years.

   ii. following each in-service inspection, the containment airlocks, the steam generator inspection/maintenance penetration, and the equipment hatch are leak tested prior to returning the plant to operation, and

   iii. any repair, replacement, or modification of a containment barrier resulting from the inservice inspections shall be followed by the appropriate leakage test.

shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion.

c. If it is determined that the leakage through a mini-purge supply and exhaust line is greater than 0.05 La an engineering evaluation shall be performed and plans for corrective action developed.

4.4.2.4 Test Frequency

a. Except as specified in b. and c. below, individual penetrations and containment isolation valves shall be tested during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years.

b. The containment equipment hatch, fuel transfer tube, steam generator inspection/maintenance penetration, and shutdown purge system flanges shall be tested at each refueling shutdown or after each use, if that be sooner.

c. The containment air locks shall be tested at intervals of no more than six months by pressurizing the space between the air lock doors. In addition, following opening of the air lock door during the interval, a test shall be performed by pressurizing between the dual seals of each door opened, within 48 hours of the opening, unless the reactor was in the cold shutdown condition at the time of the opening or has been subsequently brought to the cold shutdown condition. A test shall also be performed by pressurizing between the dual seals of each door within 48 hours of leaving the cold shutdown condition, unless the doors have not been open since the last test performed either by pressurizing the space between the air lock doors or by pressurizing between the dual door seals.

the tendon containing 6 broken wires) shall be inspected. The accepted 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 <u>Pre-Stress Confirmation Test</u>

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 <u>Containment Isolation Valves</u>

4.4.5.1 Each containment isolation valve 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 <u>Containment Isolation Response</u>

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 each containment isolation valve shall be demonstrated to be within its limit at least once per 18 months. The response time includes only the valve travel time for those valves which the safety analysis assumptions take credit for a change in valve position in response to a containment isolation signal.

ATTACHMENT B

The purpose of this amendment is to remove Table 3.6-1, "Containment Isolation Valves", from the R.E. Ginna Technical Specifications. The reference to Table 3.6-1 in Technical Specification sections 3.6.3.1, 4.4.5.1, and 4.4.6.2 will be deleted. A reference to UFSAR Table 6.2-13 will be provided in the Basis for Technical Specification 3.6. In addition, the inoperability definition and action required statement for Technical Specification 3.6.3.1 will be clarified. Technical Specifications 4.4.1.5, section a (ii) and 4.4.2.4, section b, will be revised to include the modified steam generator inspection/maintenance penetration. Technical Specification 4.4.1.5, section a (ii) will also be clarified. The temporary notes associated with the purge system and mini-purge valves (Technical Specifications 3.6.5, 4.4.2.4 section a, and 4.4.2.4 section d) will be removed since the valves have been installed. Also, the acceptance criteria for containment leakage criteria in Technical Specification 4.4.1.4 will be clarified.

The 1988 Inservice Test (IST) Program provided a complete review of the Ginna Containment Isolation Valves and their testing requirements. The information obtained during this review was submitted to the NRC to define the IST requirements for the third ten-year interval at Ginna. This submittal was subsequently approved by the NRC. As a result of this submittal and approval, numerous clarifications were required of Technical Specification Table 3.6-1 and UFSAR Table 6.2-13. However, this amendment will remove Technical Specification Table 3.6-1. The necessary changes to UFSAR Table 6.2-13 have been completed. Attachment C contains the safety evaluation related to these changes for your information.

The removal of Table 3.6-1 from the Technical Specifications and the incorporation of the required information into Ginna UFSAR Table 6.2-13 will keep the listing of the Containment Isolation Valves within a licensee controlled document. Changes to this document can only be performed under the criteria of 10CFR50.59 to ensure that no unreviewed safety questions are related to the change. Any additional changes to UFSAR Table 6.2-13 will be submitted as part of the annual UFSAR update. In addition, a report summary of the changes to the Ginna UFSAR are furnished to the NRC on an annual basis. A reference to Ginna UFSAR Table 6.2-13 is now provided in the Basis for Technical Specification 3.6 consistent with Standard Technical Specifications.

The addition of the steam generator inspection/maintenance penetration to both the UFSAR Table and the necessary Technical Specification surveillance requirements is the result of a modification to enhance containment closure during mid-loop operation (Generic Letter 88-17). No new containment isolation valves were added as a result of this modification. The addition of this penetration to the UFSAR Table and Technical Specifications 4.4.1.5, section a (ii) and 4.4.2.4, section b, causes the new penetration to be treated consistent with respect to the Personnel and Equipment Hatches, and the fuel transfer tube (see letter from R.C. Mecredy, RG&E, to A.R. Johnson, NRC, dated March 13, 1990).

The first line of Technical Specification 4.4.1.5, section a (ii) will also be modified to state "following each in-service inspection..." The hyphenation of "in-service" is to correct a typographical error only. The replacement of "one" with "each" provides greater understanding of the test frequency requirements. These changes are a minor clarification only and do not involve a technical change.

The changes related to the inoperability definition for containment isolation valves in Technical Specification 3.6.3.1 do not involve any technical changes. Instead, these clarifications will provide consistency with 10CFR50 Appendix J requirements. Changes with respect to the required actions of Technical Specification 3.6.3.1 allow consistency with Standard Technical Specifications. However, "isolation boundary" was used in place of "isolation valve" since not all penetrations have two containment isolation valves in place. For example, penetrations under the specifications for General Design Criteria 57 only require a single isolation valve; the piping provides an additional boundary. The use of "isolation boundary" is a also consistent with the column headings of the current Containment Isolation Valve Table 3.6-1. Information on what qualifies as an "isolation boundary" is provided in the Basis for Technical Specification 3.6. These criteria are consistent with the necessary General Design Criteria or exemption as appropriate.

The changes with respect to containment leakage criteria in Technical Specification 4.4.1.4 are clarifications only. All terms contained in the definition for Lt will be specified in the Technical Specifications consistent with 10CFR50 Appendix J.

The temporary notes associated with the purge and mini-purge valves in Technical Specifications 3.6.5, 4.4.2.4 section a, and 4.4.2.4 section d will be removed since these valves have been installed. This is not a technical change since the notes were only intended to be applicable until the completion of the necessary modifications.

In accordance with 10CFR50.91, these changes to the Technical Specifications have been evaluated to determine if the operation of the facility in accordance with the proposed amendment would:

1. involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. create the possibility of a new or different kind of accident previously evaluated; or

3. involve a significant reduction in a margin of safety.

These proposed changes do not increase the probability or consequences of a previously evaluated accident or create a new or different type of accident. Furthermore, there is no reduction in the margin of safety for any particular Technical Specification. The detailed changes are described in Table 1.

Therefore, Rochester Gas and Electric submits that the issues associated with this Amendment request are outside the criteria of 10CFR50.91; and a no significant hazards finding is warranted.

Attachment C

(UFSAR Changes Safety Evaluation)