August 30, 1993

Docket No. 50-244

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

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Dear Dr. Mecredy:

SUBJECT: ISSUANCE OF AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M77849)

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated October 15, 1990, as supplemented March 8, 1991, November 30, 1992, and July 13, 1993.

The amendment would remove Containment Isolation Valve Table 3.6-1 from the Technical Specifications (TS). Your November 30, 1992, letter also requested exemptions from certain provisions of 10 CFR Part 50, Appendix J. The Appendix J exemptions are independent of the changes to the TS addressed herein and will be addressed at a later date.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notice.

Sincerely,
Original signed by
Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 54 to License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

*Previously concurred

| OFFICE | PDI-3:LA | PDI-3:PM | PDI-3:PM | *OGC | PDI-3:D |
|---|---|---|---|---|---|
| NAME | SLittle | AChu:dt | AJohnson | EHoller | WButler |
| DATE | 8/30/93 | 8/31/93 | 8/13/93 | 08/13/93 | 8/13/93 |

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 30, 1993

Docket No. 50-244

Dr. Robert C. Mecredy
Vice President, Nuclear Production
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

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Sincerely,

Allen R. Johnson

Allen R. Johnson, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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   License No. DPR-18
2. Safety Evaluation

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See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Rochester Gas and Electric Corporation (the licensee) dated October 15, 1990, as supplemented March 8, 1991, November 30, 1992, and July 13, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR- 18 is hereby amended to read as follows:

(2). Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

| Remove | Insert |
|---|---|
| 3.6-1 | 3.6-1 |
| 3.6-2 | 3.6-2 |
| 3.6-3 | 3.6-3 |
| 3.6-4 | 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 | ----- |
| 3.8-1 | 3.8-1 |
| 3.8-3 | 3.8-3 |
| 3.8-5 | 3.8-5 |
| ----- | 3.8-6 |
| 4.4-4 | 4.4-4 |
| 4.4-6 | 4.4-6 |
| 4.4-7 | 4.4-7 |
| 4.4-8 | 4.4-8 |
| 4.4-11 | 4.4-11 |
| 4.4-13 | 4.4-13 |
| 4.4-14 | 4.4-14 |
| 4.4-17 | 4.4-17 |

3.6 Containment System

Applicability:

Applies to the integrity of reactor containment.

Objective:

To define the operating status of the reactor containment for plant operation.

Specification:

3.6.1 Containment Integrity

a. Except as allowed by 3.6.3, containment integrity shall not be violated unless the reactor is in the cold shutdown condition. Closed valves may be opened on an intermittent basis under administrative control.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless the boron concentration is greater than 2000 ppm.

c. Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact unless the boron concentration is greater than 2000 ppm.

3.6.2 Internal Pressure

If the internal pressure exceeds 1 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within 24 hours or the reactor rendered subcritical.

3.6.3 Containment Isolation Boundaries

3.6.3.1 With a containment isolation boundary inoperable for one or more containment penetrations, either:

a. Restore each inoperable boundary 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, one closed manual valve, or a blind flange, or

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

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%$\Delta$k/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.

In order to minimize containment leakage during a design basis accident involving a significant fission product release, penetrations not required for accident mitigation are provided with isolation boundaries. These isolation boundaries consist of either passive devices or active automatic valves and are listed in a procedure under the control of Technical Specification 6.8. Closed manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges and closed systems are considered passive devices. Automatic isolation valves designed to close following an accident without operator action, are considered active devices. Two isolation devices are provided for each mechanical penetration, such that no single credible failure or malfunction of an active component can cause a loss of isolation, or result in a leakage rate that exceeds limits assumed in the safety analyses[3].

In the event that one isolation boundary is inoperable, the affected penetration must be isolated with at least one boundary that is not affected by a single active failure. Isolation boundaries that meet this criterion are a closed and deactivated automatic containment isolation valve, a closed manual valve, or a blind flange.

The opening of closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an individual qualified in accordance with station procedures, who is in constant communication with the control room, at the valve controls, (2) instructing this individual to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to isolate the boundary and that this action will prevent the release of radioactivity outside the containment.

References:


(1) Westinghouse Analysis, "Report for the BAST Concentration Reduction for R. E. Ginna", August 1985, submitted via Application for Amendment to the Operating License in a letter from R.W. Kober, RG&E to H.A. Denton, NRC, dated October 16, 1985

(2) UFSAR - Section 3.8.1.2.2

(3) UFSAR - Section 6.2.4

3.8 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Objective

To ensure that no incident could occur during refueling operations that would affect public health and safety

Specification

3.8.1 During refueling operations the following conditions shall be satisfied.

a. Containment penetrations shall be in the following status:

   i. The equipment hatch shall be in place with at least one access door closed, or the closure plate that restricts air flow from containment shall be in place,

   ii. At least one access door in the personnel air lock shall be closed, and

   iii. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

      1. Closed by an isolation valve, blind flange, or manual valve, or

      2. Be capable of being closed by an OPERABLE automatic shutdown purge or mini-purge valve.

b. Radiation levels in the containment shall be monitored continuously.

c. Core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment and control room available whenever core geometry is being changed. When core geometry is not being changed at flange. If this condition is not met, all operations involving movement of fuel or control rods in the reactor vessel shall be suspended.

3.8.2 If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease; work shall be initiated to correct the violated conditions so that the specified limits are met; no operations which may increase the reactivity of the core shall be made.

3.8.3 If the conditions of 3.8.1.d are not met, then in addition to the requirements of 3.8.2, isolate the shutdown purge and mini-purge penetrations within 4 hours.

Basis:

The equipment and general procedures to be utilized during refueling are discussed in the UFSAR. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time. In addition, interlocks on the auxiliary building crane will prevent the trolley from being moved over stored racks containing spent fuel.

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

The analysis(3) for a fuel handling accident inside containment establishes acceptable offsite limiting doses following rupture of all rods of an assembly operated at peak power. No credit is taken for containment isolation or effluent filtration prior to release. Requiring closure of penetrations which provide direct access from containment atmosphere to the outside atmosphere establishes additional margin for the fuel handling accident and establishes a seismic envelope to protect against the potential consequences of seismic events during refueling. Isolation of these penetrations may be achieved by an OPERABLE shutdown purge or mini-purge valve, blind flange, or isolation valve. An OPERABLE shutdown purge or mini-purge valve is capable of being automatically isolated by R11 or R12. Penetrations which do not provide direct access from containment atmosphere to the outside atmosphere support containment integrity by either a closed system, necessary isolation valves, or a material which can provide a temporary ventilation barrier, at atmospheric pressure, for the containment penetrations during fuel movement.

References

(1) UFSAR Sections 9.1.4.4 and 9.1.4.5

(2) Reload Transient Safety Report, Cycle 14

(3) UFSAR Section 15.7.3.3

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.

b. The local leakage rate shall be measured for each of the following components:

   i. Containment penetrations that employ resilient seals, gaskets, or sealant compounds, piping penetrations with expansion bellows and electrical penetrations with flexible metal seal assemblies.

   ii. Air lock and equipment door seals.

   iii. Fuel transfer tube.

   iv. Isolation valves on the testable fluid systems lines penetrating the containment.

   v. Other containment components, which require leak repair in order to meet the acceptance criterion for any integrated leakage rate test.

4.4.2.2 Acceptance Criterion

Containment isolation boundaries are inoperable from a leakage standpoint when the demonstrated leakage of a single boundary or cumulative total leakage of all boundaries is greater than 0.60 La.

4.4.2.3 Corrective Action

a. If at any time it is determined that the total leakage from all penetrations and isolation boundaries exceeds 0.60 La, repairs shall be initiated immediately.

b. If repairs are not completed and conformance to the acceptance criterion of 4.4.2.2 is not demonstrated within 48 hours, the reactor shall be 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 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 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 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 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.

The Specification also allows for possible deterioration of the leakage rate between tests, by requiring that the total measured leakage rate be only 75% of the maximum allowable leakage rate.

The duration and methods for the integrated leakage rate test established by ANSI N45.4-1972 provide a minimum level of accuracy and allow for daily cyclic variation in temperature and thermal radiation. The frequency of the integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns. Refueling shutdowns are scheduled at approximately one year intervals.

The specified frequency of integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the use of weld channels to test the leaktightness of the welds during erection, (b) conformance of the complete containment to a 0.1% per day leak rate at 60 psig during preoperational testing, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60 La) of the total leakage that is specified as acceptable. Third is the tendon stress surveillance program, which provides assurance than an important part of the structural integrity of the containment is maintained.

The basis for specification of a total leakage of 0.60 La from penetrations and isolation boundaries is that only a portion of the allowable integrated leakage rate should be from those sources in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. Because most leakage during an integrated leak rate test occurs though penetrations and isolation valves, and because for most penetrations and isolation valves a smaller leakage rate would result from an integrated leak test than from a local test, adequate assurance of maintaining the integrated leakage rate within the specified limits is provided. The limiting leakage rates from the Recirculation Heat Removal Systems are judgement values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test

The pre-stress confirmation test provides a direct measure of the load-carrying capability of the tendon.

If the surveillance program indicates by extensive wire breakage or tendon stress relation that the pre-stressing tendons are not behaving as expected, the situation will be evaluated immediately. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. The containment is provided with two readily removable tendons that might be useful to such a study. In addition, there are 40 tendons, each containing a removable wire which will be used to monitor for possible corrosion effects.

Operability of the containment isolation boundaries ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Performance of cycling tests and verification of isolation times associated with automatic containment isolation valves is covered by the Pump and Valve Test Program. Compliance with Appendix J to 10 CFR 50 is addressed under local leak testing requirements.

References:


(1) UFSAR Section 3.1.2.2.7

(2) UFSAR Section 6.2.6.1

(3) UFSAR Section 15.6.4.3

(4) UFSAR Section 6.3.3.8

(5) UFSAR Table 15.6-9

(6) FSAR Page 5.1.2-28

(7) North-American-Rockwell Report 550-x-32, Autonetics Reliability Handbook, February 1963.

(8) FSAR Page 5.1-28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

## 1.0 INTRODUCTION

By letter dated October 25, 1990, as supplemented March 8, 1991, November 30, 1992, and July 13, 1993, the Rochester Gas & Electric Company (the licensee) submitted an application to amend the facility Technical Specifications (TS) relating to containment isolation and containment integrity. The July 13, 1993, application was a resubmittal of earlier applications dated October 15, 1990, March 8, 1991, and November 30, 1992. The resubmitted application of July 13, 1993, was revised to reflect changes resulting from three staff requests for additional information (RAI) dated November 12, 1990, September 26, 1991, and March 11, 1993. On June 2, 1993, the licensee provided a draft response to the March 11, 1993, RAI, and incorporated this information into this July 13, 1993, submittal. The November 30, 1992, letter also requested exemptions from certain provisions of 10 CFR Part 50, Appendix J.

This evaluation addresses the proposed TS changes. The Appendix J exemption requests will be the subject of separate staff evaluations. Each proposed TS change is discussed and evaluated below.

## 2.0 DISCUSSION AND EVALUATION

### 2.1 TS 3.6.1 - Containment Integrity

Proposed Change: TS 3.6.1 "Containment Integrity" would be amended to include a statement that "Closed valves may be opened on an intermittent basis under administrative control." The change would allow for temporary opening of locked or sealed closed isolation valves."

Staff Evaluation: Addition of the statement "Closed valves may be opened on an intermittent basis under administrative control" to the limiting conditions for operation (LCO) is part of the guidance of Generic Letter (GL) 91-08. In the GL, the staff concluded that addition of this statement to the LCO is an acceptable alternative to identifying specific valves that may be opened under administrative control. This change is acceptable based on consistency with the GL.

### 2.2 TS 3.6.3 - Containment Isolation Valve Operability

2.2.1 Deletion of Table 3.6.1

Proposed Change: Table 3.6-1 "Table of Isolation Valves," would be removed from the TS. Isolation devices would be identified in a Ginna Station Procedure which is subject to the administrative controls of the TS, including review and approval by the Plant Operations Review Committee and Plant Manager. [Note: the original application proposed to relocate the isolation valve table to the updated safety analysis report. The licensee revised its application, in response to the staff's March 11, 1993, RAI to relocate the table to a station procedure.]

Staff Evaluation: GL 91-08 provides the staff's guidance for removal of component lists from TS. The guidance states that the TS Table of Containment Isolation Valves may be relocated to a plant procedure subject to the administrative controls specified in the TS. The guidance also discusses the treatment of footnotes to the TS Table of Containment Isolation Valves. The licensee's proposed change complies with the guidance of the GL and is therefore acceptable.

2.2.2 TS 3.6.3 - Containment Isolation "Boundaries"

Proposed Change: TS 3.6.3 would be changed to apply to containment isolation "boundaries," as opposed to "valves."

Staff Evaluation: The proposed change would clarify the fact that devices other than valves may be used as acceptable containment piping penetration isolation barriers. This change is therefore acceptable.

2.2.3 Appendix J Leakage Limits as Isolation Valve Operability Criteria

Proposed Change: The statement "Isolation valves are inoperable from a leakage standpoint if the leakage is greater than that allowed by 10 CFR 50, Appendix J" would be deleted from TS 3.6.3.

Staff Evaluation: Title 10 of the Code of Federal Regulations Part 50, Appendix J does not specify leak rate limits for individual containment isolation valves or piping penetrations. Elimination of the statement will have no effect on isolation valve operability requirements and is therefore acceptable. TS 4.4.2.2 (see below) specifies the Appendix J limit on total leakage as an acceptance criterion for containment integrity. The proposed change is therefore acceptable.

2.4 TS 3.8 - Refueling Requirements

2.4.1 Containment Integrity Requirements for Refueling

Proposed Change: The TS 3.8.1 would be changed to eliminate requirements that, during refueling operations: (1) all automatic containment isolation valves be operable, or at least one valve in each line be locked closed and

(2) the 48-inch shutdown purge valves be operable or closed, or the associated flange installed. The proposed change would replace these requirements with a requirement that each penetration providing direct access from the containment atmosphere to the outside atmosphere be closed by an isolation valve, blind flange or manual valve, or be capable of being closed by an operable automatic shutdown purge or mini-purge valve. The proposed change would eliminate containment isolation operability requirements during refueling for piping that does not provide an open path between the containment atmosphere and outside environment. The licensee's application states that this is acceptable based on: (1) a fuel handling accident would not pressurize containment, and (2) the fuel handling accident analysis (for non-seismic conditions) indicates that Part 100 dose consequences acceptance criteria can be met without credit for containment isolation or filtration of effluent. The licensee also states that this is more nearly consistent with the Standard Technical Specifications.

Also included in the change are editorial changes relating to air locks, equipment hatches, and access doors. The operability requirements relating to these would not be affected.

Staff Evaluation: There is no staff position requiring primary containment integrity during refueling mode operation. During refueling (MODE 6 operation), the potential for containment pressurization as a result of an accident is greatly reduced; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. Such requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment.

The proposed piping penetration containment closure requirements would ensure that all potential piping escape paths from the containment atmosphere to the environs are closed or capable of being closed. The changes to the piping penetration TS are therefore acceptable.

2.4.2 Action Statement for Inoperable Residual Heat Removal (RHR) During Refueling

Proposed Change: TS 3.8.3 would be changed to specify that, during refueling, in the event at least one RHR pump is not operable, the licensee must isolate all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. The licensee proposes to revise TS 3.8.3 to make this action apply only to the shutdown purge and mini-purge penetrations.

Staff Evaluation: The proposed change to TS 3.8.3 is consistent with the proposed revision of TS 3.8.1. Under the proposed 3.8.1 all piping paths from the containment atmosphere to the outside environment except for the shutdown purge and mini-purge penetrations will already be closed for refueling. In the event of no operable RHR loop, it will thus only be necessary to isolate the shutdown purge and mini-purge penetrations. The proposed change is therefore acceptable.

2.5 TS 4.4.1.4 - Integrated Leak Rate Test - Definitions

Proposed Change: Expanded definitions would be provided in TS 4.4.1.4 for $P_t$, $P_a$, $L_{tm}$, $L_a$ and $L_{am}$.

Staff Evaluation: The licensee's safety evaluation accompanying the application states that the change is for clarification only, and that the definitions are consistent with existing requirements for Ginna and do not constitute a technical change. The staff has reviewed these changes and agrees. The proposed changes are therefore acceptable.

2.6 TS 4.4.1.5 - Leak Rate Test Frequency For Steam Generator Inspection/Maintenance Penetration

Proposed Change: TS 4.4.1.5.a.ii would be modified to require a local leak test of the generator inspection/maintenance penetration prior to returning the plant to operation after each inservice inspection.

Staff Evaluation: The proposed change reflects the conversion of a spare, previously capped, 10-inch containment penetration into a new, blind-flange penetration which is used for steam generator maintenance during outages. The new penetration facilitates cabling of maintenance equipment in a manner that would enable more rapid establishment of containment integrity in the event of a mid-loop event. The inclusion of the new containment penetration to the list of penetrations to be tested prior to return of the plant to operation is similar to the treatment of personnel and equipment hatches and the fuel transfer tube. This surveillance requirement, plus the proposed change to TS 4.4.2.4 described below ensure integrity of the new penetration during plant operation and is acceptable.

2.7 TS 4.4.2.2 - Acceptance Criteria for Local Leak Rate Tests

Proposed Change: TS 4.4.2.2 presently states that "the total leakage from all penetrations and isolation valves shall not exceed 0.60 $L_a$." The proposed change would revise the statement to read "Containment isolation boundaries are inoperable from a leakage standpoint when the demonstrated leakage of a single boundary or the cumulative total leakage of all boundaries is greater than 0.60 $L_a$."

Staff Evaluation: The proposed change clarifies that leak tightness is an operability criterion for all containment boundaries and replaces the term "isolation valve" with "isolation boundary." The proposed change would provide consistency and clarification and does not affect operability or surveillance requirements. It is therefore acceptable.

2.8 TS 4.4.2.3 - Corrective Action for Leak Rate Tests

Proposed Change: The action requirement for failure of "penetration and isolation valves" to meet the 0.60 $L_a$ acceptance criterion would be clarified as applicable to "penetrations and isolation boundaries."

Staff Evaluation: The proposed change is a simple clarification that does not affect operability or surveillance requirements and is therefore acceptable.

2.9 TS 4.4.2.4.a - Test Frequency for Mini-Purge Valves

Proposed Change: A statement requiring that the four mini-purge isolation valves shall be tested at 6-month intervals for 2 years following installation of the mini-purge system would be deleted from the surveillance requirement regarding local leak rate test frequencies.

Staff Evaluation: Deletion of these requirements from the TS is acceptable since they have expired. The intent of the original accelerated leak test rate was to gather increased data regarding the leak tightness of the valves under service conditions to determine if increased testing is needed on a permanent basis. This relates to staff concerns about large isolation valves having resilient seats. The licensee and staff reviewed the test history of the valves and confirmed that a 2-year test interval is supported by valve-specific historical test results.

2.10 TS 4.4.2.4.d - Containment Purge Isolation Valve Testing

Proposed change: Text related to a former temporary requirement for increased test frequency of the purge valves, pending provision of flanges, would be deleted.

Staff Evaluation: The expired text may be deleted from the TS. The shutdown flanges have been provided and are currently tested after each refueling shutdown or use under the requirements of TS 4.4.2.4.b. This change is therefore acceptable.

2.11 TS 4.4.5.1 - IST Requirements for Isolation Valve Operability

Proposed Change: A reference to the TS Table of Isolation Valves/Table 3.6.1, would be deleted in the isolation valve surveillance requirements relating to 10 CFR 50.55a inservice testing program.

Staff Evaluation: Deletion of the table reference would reflect the removal of the table from the TS as discussed in 2.2 above, and is acceptable based on the guidance of GL 91-08.

2.12 TS 4.4.6.2 - Response Time Testing for Isolation Valves

2.12.1 Reference to Table 3.6.1

Proposed Change: A reference to the TS Table of Isolation Valves/Table 3.6.1, would be deleted in the isolation valve response time testing requirement.

Staff Evaluation: Deletion of the table reference would reflect the removal of the table from the TS as discussed in 2.2 above, and is acceptable based on the guidance of GL 91-08.

2.12.2 Limitation on Scope of Valves Requiring Response Time Testing

Proposed Change: The containment isolation valve response time testing requirement would be limited to valves for which a change of position in response to a containment isolation signal is assumed in the accident analyses.

Staff Evaluation: The proposed change is consistent with guidance provided by the staff in a November 12, 1990, RAI. In that correspondence, the staff specifically recommended use of the wording now proposed in order to avoid use of undefined terms and to more explicitly identify which valves require response time testing. Limiting response time surveillance testing to containment isolation valves that change position in response to a containment isolation signal is consistent with the assumptions used in analyses of the radiological consequences of design basis accidents and is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (55 FR 51186). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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