50-244

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

February 6, 1995

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

SUBJECT: ISSUANCE OF AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M89473)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 58 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in partial response to your application dated May 13, 1994, as supplemented by letters of June 24, and September 27, 1994, in which you proposed to amend Appendix A of Operating License DPR-18 to revise Section 6.0 "Administrative Controls" of the Ginna Technical Specifications (TSs).

The September 27, 1994, letter would change the title of Senior Vice President, Production and Engineering, include a provision to allow future title changes without license amendment, and implement those changes in NUREG-1431 "Standard Technical Specification - Westinghouse Plants," dated September 1992, by relocating to licensee controlled documents those specifications controlled by regulations and the review and audit requirements. The NRC staff has reviewed the partial amendment proposal in response to your September 27, 1994, application and found it acceptable. The enclosed safety evaluation (SE) documents the basis for the staff's conclusion.

All other changes proposed in the May 13 and June 24, 1994, letters have been deferred until Rochester Gas and Electric Corporation's (RG&E's) proposed improved TSs for conversion to NUREG-1431 are submitted at a later date.

The NRC staff may perform an audit of these relocated requirements after final completion of the improved TS for conversion to NUREG-1431, to assure that an appropriate level of control has been achieved. The staff encourages RG&E to perform its own audit to ensure that the commitments specified in the May 13, June 24, and September 27, 1994, letters, and this SE have been fully accomplished.

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DF01

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly Federal Register 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

Docket No. 50-244

Enclosures: 1. Amendment No. 58 to License No. DPR-18
2. Safety Evaluation

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A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Allen R. Johnson

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

Docket No. 50-244

Enclosures: 1. Amendment No. 58 to License No. DPR-18
2. Safety Evaluation

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. 58
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 May 13, 1994, as supplemented June 24 and September 27, 1994, 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. 58, 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, or upon NRC approval and implementation of the licensee's Quality Assurance Plan, Revision 20, whichever is later.

FOR THE NUCLEAR REGULATORY COMMISSION

A E Edison for

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: February 6, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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
ii	ii
3.1-21	3.1-21
3.5-2a	3.5-2a
3.6-3	3.6-3
3.16-2	3.16-2
6.2-1	6.2-1
6.2-2	6.2-2
6.2-3	6.2-3
6.2-4	----
6.4-1	6.4-1
6.5-1	6.5-1
6.5-2	----
6.5-3	----
6.5-4	----
6.5-4a	----
6.5-5	----
6.5-6	----
6.5-7	----
6.5-8	----
6.5-8a	----
6.5-9	----
6.5-10	----
6.5-11	----
6.5-12	
6.6-1	6.6-1
6.7-1	6.7-1
6.8-1	6.8-1
6.8-2	----
6.9-4	6.9-4
6.9-6	6.9-6
6.9-7	6.9-7
6.10-1	6.10-1
6.10-2	----
6.10-3	----
6.11-1	6.11-1
6.13-1	6.13-1
6.13-2	6.13-2
6.15-1	6.15-1
6.16-1	6.16-1
6.17-1	6.17-1

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3.1.4 Maximum Coolant Activity

Specifications

3.1.4.1 Whenever the reactor is critical or the reactor coolant average temperature is greater than 500°F:

a. The total specific activity of the reactor coolant shall not exceed $84/\overline{E}$ μCi/gm, where $\overline{E}$ is the average beta and gamma energies per disintegration in Mev.

b. The I-131 equivalent of the iodine activity in the reactor coolant shall not exceed 0.2 μCi/gm.

c. The I-131 equivalent of the iodine activity on the secondary side of a steam generator shall not exceed 0.1 μCi/gm.

3.1.4.2 If the limit of 3.1.4.1.a is exceeded, then be subcritical with reactor coolant average temperature less than 500°F within 8 hours.

3.1.4.3 a. If the I-131 equivalent activity in the reactor coolant exceeds the limit of 3.1.4.1.b but is less than the allowable limit shown on Figure 3.1.4-1, operation may continue for up to 168 hours.

3.5.5.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

(i) the setpoint shall be immediately corrected without declaring the channels inoperable; <u>or</u>

(ii) immediately suspend the release of effluents monitored by the effected channel; <u>or</u>

(iii) declare the channel inoperable.

3.5.5.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-5. Exert best efforts to return the instruments to OPERABLE status within 31 days and, if unsuccessful, explain in the next Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.6 Control Room HVAC Detection Systems

3.5.6.1 During all modes of plant operation, detection systems for chlorine gas, ammonia gas and radioactivity in the control room HVAC intake shall be operable with setpoints to isolate air intake adjusted as follows:

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.

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 the Quality Assurance Program. 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.

6.9-2 when averaged over any calendar quarter, a Special Report shall be submitted to the Commission within thirty days which includes an evaluation of any release conditions, environmental factors or other aspects which caused the reporting levels of Table 6.9-2 to be exceeded.

When more than one of the radionuclides in Table 6.9-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 6.9-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is greater than the calendar year limit of Specifications 3.9.1.2.a or 3.9.2.2.b. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3.16.1.4 If milk or fresh leafy vegetable samples are unavailable for more than one sample period from one or more of the sampling locations indicated by the ODCM, a discussion shall be included in the Radioactive Effluent Release Report which identifies the cause of the unavailability of samples and identifies locations for

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organization

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all Plant management positions. Those relationships shall be documented and updated, as appropriate, in the form of organization charts. These organization charts will be documented in the UFSAR and updated in accordance with 10 CFR 50.71.

b. The Senior Vice President, Customer Operations*, shall have corporate responsibility for overall Plant nuclear safety, and shall take any measures needed to assure acceptable performance of the staff in operating, maintaining, and providing technical support in the Plant so that continued nuclear safety is assured.

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* An alternate title may be designated for this position in accordance with 10 CFR 50.54(a)(3). All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

c. The Plant Manager, Ginna Station shall have responsibility for overall unit operation and shall have control over those resources necessary for safe operation and maintenance of the Plant.

d. The persons responsible for the training, health physics and quality assurance functions may report to an appropriate manager onsite, but shall have direct access to responsible corporate management at a level where action appropriate to the mitigation of training, health physics and quality assurance concerns can be accomplished.

6.2.2 Facility Staff

The Facility organization shall include the following:

a. An auxiliary operator shall be assigned to the shift crew with fuel in the reactor. An additional auxiliary operator shall be assigned to the shift crew above Cold Shutdown.

b. At least one licensed operator shall be present in the control room when fuel is in the reactor. In addition, above Cold Shutdown, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.

c. Shift crew composition may be less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 6.2.2.a and 6.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore shift crew composition to within the minimum requirement.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. Adequate shift coverage shall be maintained without routine heavy use of overtime. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions including senior reactor operators, reactor operators, health physicists*, auxiliary operators, and key maintenance personnel. Changes to the guidelines for the administrative procedures shall be submitted to the NRC for review.

f. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be assigned to the shift crew above Cold Shutdown.

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* An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Division Training Manager* and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55.

6.4.2 The training program shall meet or exceed NFPA No. 27, 1975 Section 40, except that (1) training for salvage operations need not be provided and (2) the Fire Brigade training sessions shall be held at least quarterly. Drills are considered to be training sessions.

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* An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

6.5 (Deleted)

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6.6 (Deleted)

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6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The provisions of 10 CFR 50.36(c)(1)(i)(A) shall be complied with immediately.

b. The Safety Limit violation shall be reported to the Senior Vice President, Customer Operations*, to the offsite review function, and to the NRC immediately.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the onsite review function. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the NRC, the offsite review function, and the Senior Vice President, Customer Operations* within two weeks of the violation.

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* An alternate title may be designated for this position in accordance with 10 CFR 50.54(a)(3). All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

b. Fire Protection Program implementation.

c. The radiological environmental monitoring program.

d. Offsite Dose Calculation Manual implementation.

e. Process Control Program implementation.

and directions from the reactor, and the results of the participation in an interlaboratory comparison program.

6.9.1.4 Radioactive Effluent Release Report

Routine radioactive effluent release reports covering the operation of the unit during the previous twelve months of operation shall be submitted by May 1 of each year. This report shall include a summary, on a quarterly basis, of the quantities of radioactive liquid and gaseous effluents and solid waste released as outlined in Regulatory Guide 1.21, Revision 1.

This report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each of the previous four calendar quarters as outlined in Regulatory Guide 1.21, Revision 1. In addition, the site boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM. This same report shall include an annual summary of hourly meteorological data collected over the previous calendar year. Alternatively, the licensee has the option of retaining this summary on site in a file that shall be provided to the NRC upon request.

Also, the report shall include any nearby location(s) identified by the land use census which

6.9.2 Unique Reporting Requirements

6.9.2.1 Annually: Results of required leak test performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

6.9.2.2 Annually: A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. (NOTE: This tabulation supplements the requirements of Section 20.407 of 10CFR Part 20)

6.9.2.3 (Deleted)

6.9.2.4 Reactor Overpressure Protection System Operation

In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission within thirty days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any other corrective action necessary to prevent recurrence.

6.10 (Deleted)

(Intentionally Left Blank)

6.11 (Deleted)

(Intentionally Left Blank)

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20:

a. Each High Radiation Area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit* (RWP). Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

(1) A radiation monitoring device which continuously indicates the radiation dose rate in the area.

(2) A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

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* Radiation Protection** personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, providing they are following plant radiation protection procedures for entry into high radiation areas.

** An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.

(3) A Qualified health physicist* (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who will perform periodic radiation surveillance at the frequency specified in the HPWP. The surveillance frequency will be established by a plant Health Physicist*.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1 a. above, and in addition locked doors shall be provided to prevent unauthorized entry into these areas and the keys to unlock these locked doors shall be maintained under the administrative control of the Shift Supervisor on duty.

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* **An alternate title may be designated for this position. All requirements of these Technical Specifications apply to the position with the alternate title as apply with the specified title. Alternate titles shall be specified in the Updated Final Safety Analysis Report.**

6.15 Offsite Dose Calculation Manual (ODCM)

6.15.1 Any changes to the ODCM shall be made by the following method:

6.15.1.a Licensee initiated changes shall be submitted to the Commission with the Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

(i) sufficiently detailed information to support the rationale for the change.

(ii) a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

(iii) documentation of the fact that the change has been reviewed and found acceptable by the onsite review function.

6.15.1.b Licensee initiated changes shall become effective after review and acceptance by the onsite review function on a date specified by the licensee.

6.16 <u>Process Control Program</u> (PCP)

6.16.1 Any changes to the PCP shall be made by the following method:

6.16.1.a Licensee initiated changes shall be submitted to the Commission with the Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

(i) sufficiently detailed information to support the rationale for the change;

(ii) a determination that the change will not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

(iii) documentation of the fact that the change has been reviewed and found acceptable by the onsite review function.

6.16.1.b Licensee initiated changes shall become effective after review and acceptance by the onsite review function on a date specified by the licensee.

6.17 Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous and Solid)

FUNCTION

6.17.1 The radioactive waste treatment systems (liquid, gaseous and solid) are those systems defined in Technical Specification 5.5.

6.17.2 Major changes to the radioactive waste systems (liquid and gaseous) shall be reported by the following method. For the purpose of this specification, "major changes" is defined in Specification 6.17.3 below.

6.17.2.1 The Commission shall be informed of all major changes by the inclusion of a suitable discussion or by reference to a suitable discussion of each change in the Radioactive Effluent Release Report for the period in which the changes were made. The discussion of each change shall contain:

a) a summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);

b) sufficient detailed information to support the reason for the change;

c) a detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 58 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 May 13, 1994, as supplemented June 24 and September 27, 1994, Rochester Gas and Electric Corporation (RG&E), proposed to amend Appendix A of Operating License No. DPR-18 to revise Section 6.0 "Administrative Controls" of the R. E. Ginna (Ginna) Technical Specifications (TSs) consistent with NUREG-1431 "Standard Technical Specifications - Westinghouse Plants" dated September 1992. Included in RG&E's proposal to amend Appendix A were minor changes to Section 3.0 "Limiting Conditions for Operation" associated with the Section 6.0 revisions. The September 27, 1994, letter limited RG&E's previously requested TSs changes of May 13 and June 24, 1994, to those administrative controls of Section 6.0 relating to the Quality Assurance Program. The September 27, 1994, letter requested that other TSs changes, proposed in the May 13 and June 24, 1994, letter, be deferred until RG&E's proposed improved TSs for conversion to NUREG-1431 (improved Ginna TSs) are submitted at a later date.

2.0 EVALUATION

Section 50.36 of Title 10 of the Code of Federal Regulations established the regulatory requirements related to the content of TSs. The rule requires that TSs include items in specific categories, including safety limits, limiting conditions for operation, surveillance requirements and administrative controls; however, the rule does not specify the particular requirements to be included in a plant's TSs. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (58 FR 39132) to determine which of the design conditions and associated surveillances need to be located in the TSs because the requirement is "necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." Briefly, those criteria are: (1) detection of abnormal degradation of the reactor coolant pressure boundary; (2) boundary conditions for design basis accidents and transients; (3) primary success paths to prevent or mitigate design basis accidents and transients; and (4) functions determined to be important to risk or operating experience. The Commission's final policy statement acknowledged that its implementation may result in the relocation of existing TSs requirements to licensee controlled documents and programs.

Requirements that are in the administrative control section of the Ginna current TSs (CTSs), but do not meet the criteria set forth in the policy statement for inclusion in TSs or are already covered by Regulations, will be relocated to appropriate licensee controlled documents or removed from the TSs, as appropriate. These requirements are not required by 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, and do not fall within any of the four criteria set forth in the Commission's Final Policy Statement.

In general, RG&E has proposed to relocate these items to the plant-specific procedures which implement the regulations and the Quality Assurance (QA) Program. Future changes to those provisions relocated to the plant procedures and QA Program will be governed by §50.59 and §50.54 respectively. This approach ensures an auditable and appropriate control over the relocated requirements and future changes to these provisions.

In accordance with the guidance in the policy statement, RG&E has proposed to relocate or reorganize all or portions of the following CTSs to other licensee controlled documents:

CTS Section	Title
3.1.4	Maximum Coolant Activity
6.2.2.a	Minimum Shift Composition
6.2.2.d	Senior Reactor Operator (SRO) Present During Fuel Movement
Table 6.2-1	Minimum Shift Crew Composition
6.5	Review and Audit
6.6	Reportable Event Action
6.8.1.d	Security Plan Implementation
6.8.1.e	Emergency Plan Implementation
6.8.2	Review and Approval Process
6.8.3	Temporary Change Process
6.9.2.3	Specific Activity Analysis Report
6.9.2.5	Special Reports
6.10	Record Retention
6.11	Radiation Protection Program

Specific Activity Analysis Report

The licensee proposes that the requirements in CTS 3.1.4.3.a and 6.9.2.3 for the results of special activity analysis in which the primary coolant exceeded the limits of CTS 3.1.4.1.a and 3.1.4.1.b be reported to the Commission not be retained in TSs. 10 CFR 50.73(a) provides requirements for the licensee to submit a License Event Report (LER) to report fuel cladding failures that exceed expected values or that are caused by unexpected factors, i.e., being seriously degraded. The LERs will contain the same type of information required by CTS 3.1.4.3.a and 6.9.2.3. The above reporting requirements are included in the licensee procedures which implement 10 CFR 50.72 and 10 CFR 50.73. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing duplication of them from TSs is acceptable.

Minimum Shift Composition

The licensee proposes that CTS 6.2.2.a and associated Table 6.2.2-1 be deleted from TSs since 10 CFR 50.54(k), (l) and (m) provide the requirements for shift complement regarding licensed reactor operators. The regulations describe the minimum shift composition for operating modes, as well as for cold shutdown and refueling. The requirements in this specification implement 10 CFR 50.54. Additionally, the licensee proposes to reorganize CTS 6.2.2 "Facility Staff" in the format of TS 6.2.2.a, 6.2.2.b, 6.2.2.e and 6.2.2.f to specify when licensed and nonlicensed operators are required to be in the control room. The staff concludes that the regulatory requirements provide sufficient control of these provisions and removing duplication of them from TSs is acceptable.

Senior Reactor Operator (SRO) Present During Fuel Movement

The licensee proposes that the requirement in CTS 6.2.2.d that an SRO be present during fuel handling and to supervise all core alternations not be retained in the TSs. This is required by 10 CFR 50.54(m)(2)(iv) and need not be controlled by TSs to assure safe operation of the facility. The current regulation states:

> "Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person."

This requirement is specified in the licensee procedures which implement 10 CFR 50.54. The staff concludes that the regulatory requirements provide sufficient control of these provisions and removing duplication of them from TSs is acceptable.

Review and Audits

The licensee proposed that the review and audit functions specified in CTS 6.5, and 6.6.1.b be relocated from the CTS on the basis that they are adequately controlled by the QA program. Similarly, the review and audit functions for the security and emergency plans are relocated to their respective plans in accordance with Generic Letter (GL) 93-07, as described in more detail below.

The review and audit functions do not need to be controlled by TSs because an equivalent level of regulatory control can be achieved by the QA Program while providing for a more appropriate change control process. Items relocated to the QA Program are items not necessary to assure safe operation of the plant. The level of safety of plant operation is unaffected by this change, and the NRC and licensee resources associated with processing license amendments pursuant to the existing administrative controls may be used more effectively. In addition, the following considerations support relocating these items from the TSs:

- The onsite review function, composition, alternate membership, meeting frequency, quorum, responsibilities, authority and records are all covered in equivalent detail in ANSI N18.7-1976. These requirements are in the QA Program and change control is provided by 10 CFR 50.54(a).

- The offsite review group is also addressed, although with less detail, in ANSI N18.7-1976. The QA Program includes the requirements for the offsite review group. Therefore, duplicating the review and audit function of the offsite review group in the improved TSs is unnecessary.

- Audit requirements are specified in the QA Program to satisfy 10 CFR Part 50, Appendix B, Criterion XVII. Audits are also covered by ANSI N18.7, ANSI N45.2, 10 CFR 50.54(t), 10 CFR 50.54(p), and 10 CFR Part 73. Therefore, duplication of these regulatory requirements does not enhance the level of safety of the plant, nor are the provisions relating to audits necessary to assure safe operation of the facility.

- Although there are aspects of the fire detection and mitigation functions that have been determined to be risk significant, the minimum requirements for those functions are established in the regulations (§50.48), with which the licensee must comply regardless of whether the requirements are restated in the TSs. In addition, the staff has concluded that sufficient regulatory controls exist under §50.54(a) for future changes to the review and audit provisions related to implementation of the fire protection program to assure continued protection of the public health and safety.

Reportable Event Action

The licensee proposes that the requirement in CTS 6.6.1.a that the Commission be notified of all reportable events not be retained in the TSs. The Code of Federal Regulations at 10 CFR 50.73(a)(2) provides requirements for the licensee to submit a LER for all reportable events specified in 10 CFR 50.73. The reports are required to be submitted within 30 days and will contain the same type of information required by CTS 6.6.1.a. The above requirements are included in the licensee procedures which implement 10 CFR 50.72 and 10 CFR 50.73. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing duplication of them from TSs is acceptable.

Security Plan Implementation and Emergency Plan Implementation

The licensee proposes to relocate the requirements to establish, implement, and maintain procedures related to the Emergency Plan (CTS 6.8.1.e) and

Security Plan (CTS 6.8.1.d). Since the Security Plan requirements are specified in 10 CFR 50.54, 73.40, 73.55, and 73.56 and the Emergency Plan requirements are specified in 10 CFR 50.54 and 10 CFR Part 50, Appendix E, Section V, the staff in GL 93-07 recommended removal of these requirements from the Standard Technical Specifications (STS) and relocation to their respective plans.

Further changes in these review requirements must be made in accordance with 10 CFR 50.54(p) for the Security Plan and 10 CFR 50.54(q) for the Emergency Plan. The staff concludes that, in conjunction with this change to the plans, the sufficient requirements for emergency planning in 10 CFR 50.47 and 50.54 and for security in 10 CFR 50.54 and 73.55 for drills, exercises, testing, and maintenance of the program, will be met. The staff concludes that these regulatory requirements are sufficient and, therefore, removing these duplicate provisions from TSs is acceptable.

<u>Review and Approval Process and Temporary Change Process</u>

The licensee is proposing to relocate both the review and approval process (CTS 6.8.2) and the temporary change process (CTS 6.8.3) for procedures to the QA Program. This proposal is based on the existence of the following requirements which duplicate 10 CFR 50.36 in these areas.

The requirements for procedure control are addressed in Criterion II and Criterion V of Appendix B to Part 50. ANSI N18.7-1976, which is an NRC staff-endorsed document used in the development of many licensee QA plans, also contains specific requirements related to procedures. The licensee has committed to follow ANSI N18.7-1976 as a means to comply with Appendix B to Part 50. ANSI N18.7-1976, Section 5.2.2 discusses procedure adherence. This section clearly states that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. ANSI N18.7-1976 also discusses temporary changes to procedures, and requires review and approval of procedures to be defined. ANSI N18.7-1976, Section 5.2.15 describes the review, approval and control of procedures. This section describes the requirements for the licensee's QA Program to provide measures to control and coordinate the approval and issuance of documents, including changes thereto, which prescribe all activities affecting quality. The section further states that each procedure shall be reviewed and approved prior to initial use. The required reviews are also described. ANSI N45.2-1971, Section 6, also specifies that the QA Program describe procedure requirements.

The licensee has proposed to relocate those provisions for review, approval and changes to procedures, that are not otherwise covered by regulatory requirements, to the QA Program. Items relocated to the QA Program are items not necessary to assure safe operation of the plant. Future changes to the QA Program are governed by §50.54(a). The staff concludes that sufficient regulatory controls exist for the QA Program such that removing those provisions from the TSs and relocating them to the QA Program is acceptable.

Report Recipients

CTS 6.9.2.5 "Special Reports" specifies who is to receive copies of these reports. Recipients for documents sent to the NRC staff are governed by 10 CFR 50.4. This requirement is included in the licensee procedures which implement 10 CFR 50.4. The staff concludes that these regulatory requirements provide sufficient control of these provisions and removing duplication of them from the TSs is acceptable.

Record Retention

The licensee proposes that the requirements in CTS 6.10 on record retention be relocated from the CTS on the basis that they are adequately addressed by the QA Program (10 CFR Part 50, Appendix B, Criteria XVII). Items relocated to the QA Program are items not necessary to assure safe operation of the plant.

Facility operations are performed in accordance with approved written procedures. Areas controlled by procedures include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, safety-related maintenance, surveillance and testing, and radiation control. Facility records document appropriate station operations and activities. Retention of these records provides documentation retrievability for review of compliance with requirements and regulations. Post-compliance review of records does not directly assure operation of the facility in a safe manner, as activities described in these documents have already been performed. In addition, numerous other regulations such as 10 CFR Part 20, Subpart L, and 10 CFR 50.71 require the retention of certain records related to operation of the nuclear plant. The staff concludes that these regulatory requirements provide sufficient control of these recordkeeping provisions and removing them from TSs is acceptable.

Radiation Protection Program

The licensee proposes to relocate the program description in CTS 6.11 for the Radiation Protection Program. The Radiation Protection Program requires procedures to be prepared for personnel radiation protection consistent with the requirements of 10 CFR Part 20. The requirement to have procedures to implement Part 20 is also contained within 10 CFR 20.1101(b). Periodic review of these procedures is addressed under 10 CFR 20.1101(c).

The licensee has proposed to relocate these provisions to the appropriate plant procedures which implement 10 CFR Part 20 and 10 CFR 20.1101. The staff concludes that the requirements of the rule provide sufficient controls for these provisions, and that §50.59 provides adequate controls for future changes to the related plant procedures. On this basis, the staff concludes that the above requirements, and duplication of them, do not need to be controlled by TSs, and changes to the requirements are adequately controlled by 10 CFR 50.4, 10 CFR 50.54, 10 CFR 50.59, 10 CFR 50.72, 10 CFR 50.73, and 10 CFR Part 50, Appendix B.

A number of administrative changes were also proposed for the Administrative Controls as a result of the restructuring of the Administrative Controls section of the Ginna TSs. These changes, as described in more detail below, are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content. These items are relocated within the TSs itself, or duplication within the TSs is proposed. The following changes are acceptable because they are purely administrative:

1. The title of the Senior Vice President, Production and Engineering, specified in CTS 6.2.1.b, 6.7.b and 6.7.d has been revised to Senior Vice President, Customer Operations. This is an individual's title change only. The Senior Vice President continues to have corporate responsibility for overall plant nuclear safety and is chairman of the offsite review function (i.e., Nuclear Safety Audit and Review Board). The responsibilities of key organization positions, including the Senior Vice President, Production and Engineering, are specified in the Ginna Station QA Program.

   In addition, the licensee has proposed adding a footnote for the titles specified in CTS 6.2.1.b, 6.4.1, 6.7.1.b, 6.7.1.d and 6.13.1.a and associated footnote *, and new TS 6.2.2.e to preclude the need for future TS changes for title changes only by stating "All requirements of these TSs apply to the position with the alternate title as apply with the specified title." CTS 6.2.1.a requires that lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all plant management positions. These organization relationships and responsibilities are documented, specified, and updated as appropriate in the Updated Final Safety Analysis Report (UFSAR) and the QA Program; changes to the UFSAR and QA Program are controlled by §50.59 and §50.54(a)(3) respectively.

   The "alternative title" designated for a position would have all requirements of the TSs apply to the alternative title as apply to the specified title. Alternative titles would be specified in the UFSAR or the QA Program. The staff finds that the requested change is acceptable because the unit staff qualifications specified in the UFSAR and QA Program related to a specified title are also met with respect to the alternate title.

2. As a result of the relocation of CTS 6.5, 6.8.2, and 6.8.3 to the QA Program, editorial changes to account for these relocations have been made to CTS 6.7.b, 6.7.c, 6.7.d, 6.15.1, 6.16.1 and the Bases for CTS 3.6.

3. The licensee has updated CTS 6.8.1.a from Regulatory Guide (RG) 1.33, November 1972 to the latest revision which is Revision 2, February 1978. As a result of this change, CTS 6.8.1.b and 6.8.1.c can be deleted since they are specified as required by Revision 2 of RG 1.33.

4. The Code of Federal Regulations 10 CFR 50.36a(a)(2) requires that the radioactive effluent release report be submitted to the Commission annually, rather than semiannually. In order to conform to the regulation, the licensee has changed the reporting requirement of CTS 6.9.1.4 to annually, in accordance with the guidance provided in GL 89-01, "Guidance for the Implementation of Programmatic Controls in the Administrative Controls Section of Technical Specifications and Relocation of Procedural Details of Radiological Effluent Technical Specifications to Offsite Dose Calculation Manual or to the Process Control Manual," as modified by NUREG-1431. The provisions in CTS 3.5.5.3, 3.16.1.4, 6.15.1.a, 6.16.1.a, and 6.17.2.1 have been modified to reflect this change.

These provisions of the current technical specifications described above are not required by 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement.

In addition, the staff finds that sufficient regulatory controls exist under §50.59 and §50.54(a) and the other regulations set forth above to assure continued protection of the public health and safety. Accordingly, the staff has concluded that these requirements may be relocated from the TSs to the plant procedures, which are described in the FSAR, or the QA Program, as applicable. The documentation for future changes to those provisions relocated from the TSs will be maintained by RG&E in accordance with the record retention requirements specified in their QA Program.

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

This amendment changes recordkeeping, reporting, and administrative procedures and requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this 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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Date: February 6, 1995