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

April 8, 2010

Mr. John T. Carlin
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: REVISIONS TO INSERVICE TEST PROGRAM TECHNICAL SPECIFICATIONS (TAC NO. ME2269)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 110 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 18, 2009.

The amendment revises TS 5.5.7, "Inservice Testing Program," by incorporating TS Task Force Traveler (TSTF) 479, "Changes to Reflect Revision of 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.55a," and TSTF-497, "Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less." Specifically, the amendment (1) replaces references to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, with the ASME Code for Operation and Maintenance of Nuclear Power Plants for inservice testing activities, and (2) applies the extension allowance of SR 3.0.2 to other normal and accelerated inservice testing frequencies of 2 years or less that were not included in the frequencies listed in TS 5.5.7.a.

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

Sincerely,

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 110 to Renewed License No. DPR-18
2. Safety Evaluation

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

R.E. GINNA NUCLEAR POWER PLANT, LLC

DOCKET NO. 50-244

R.E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 110
Renewed 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 R.E. Ginna Nuclear Power Plant, LLC (the licensee) dated September 18, 2009, 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 Renewed 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. 110, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and Technical Specifications

Date of Issuance: April 8, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 110

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
5.5-1	5.5-1
5.5-2	5.5-2
5.5-3	5.5-3
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13

(b) Pursuant to the Act and 10 CFR Part 70, to possess and use four (4) mixed oxide fuel assemblies in accordance with the RG&E's application dated December 14, 1979 (transmitted by letter dated December 20, 1979), as supplemented February 20, 1980, and March 5, 1980;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

(1) <u>Maximum Power Level</u>

Ginna LLC is authorized to operate the facility at steady-state power levels up to a maximum of 1775 megawatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 110, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) <u>Fire Protection</u>

(a) The licensee shall implement and maintain in effect all fire protection features described in the licensee's submittals referenced in and as approved or modified by the NRC's Fire Protection Safety Evaluation (SE) dated February 14, 1979, and

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.1 Offsite Dose Calculation Manual (ODCM)

The ODCM shall contain:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by the onsite review function and the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include Containment Spray, Safety Injection, and Residual Heat Removal in the recirculation configuration. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Deleted

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I and 40 CFR 141;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 <u>Component Cyclic or Transient Limit Program</u>

This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in UFSAR Table 5.1-4 to ensure that components are maintained within the design limits.

5.5.6 <u>Pre-Stressed Concrete Containment Tendon Surveillance Program</u>

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies applicable to the ASME code for Operation and Maintenance of Nuclear Power Plants (ASME OM Code) and applicable Addenda as follows:

ASME OM Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing program for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady-state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for each SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial, and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

   1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

   2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

a. Containment Recirculation Fan Cooler System

   1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

   2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

b. Control Room Emergency Air Treatment System (CREATS)

   1. Demonstrate the pressure drop across the combined HEPA filters, the prefilters, the charcoal adsorbers and the post-filters is < 11 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 0.05%.

3. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 0.05%, when tested under ambient conditions.

4. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 1.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F), a relative humidity of 95%, and a face velocity of 61 ft/min.

c. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

5.5.11 <u>Explosive Gas and Storage Tank Radioactivity Monitoring Program</u>

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NUREG-0133.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

 1. an API gravity or an absolute specific gravity within limits,

 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

 3. a clear and bright appearance with proper color or a water and sediment content within limits; and

b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license; or

   2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71e.

5.5.14 <u>Safety Function Determination Program (SFDP)</u>

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the supported system(s) is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the May 31, 1996 Type A test shall be performed by May 31, 2011.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

5.5.16 <u>Control Room Envelope Habitability Program</u>

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Treatment System (CREATS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Licensee controlled programs that will be used to verify the integrity of the CRE boundary. Conditions that generate relevant information from these programs will be entered into the corrective action process and shall be trended and used as part of the 36 month assessments of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability and determining CRE unfiltered inleakage as required by paragraph c.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 110 TO RENEWED FACILITY

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 18, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092680060), the R.E. Ginna Nuclear Power Plant, LLC, the licensee, requested an amendment to the R.E. Ginna Nuclear Power Plant Renewed Facility Operating License No. DPR-18. The licensee proposed to revise Technical Specification (TS) 5.5.7, "Inservice Testing Program," by replacing references to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, with references to the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). The licensee also proposed to revise TS 5.5.7 by applying the extension allowance of Surveillance Requirement (SR) 3.0.2 to other normal and accelerated inservice testing (IST) frequencies of 2 years or less that were not included in the frequencies listed in TS 5.5.7.a. The proposed changes are based on the NRC's approval of Technical Specification Task Force Travelers (TSTFs) 479 and 497.

2.0 REGULATORY EVALUATION

The following documents and NRC requirements were applicable to the staff's review of the licensee's amendment request:

- Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 36(c) requires that TSs include items in five specific categories, which include the category of administrative controls. The proposed change to TS 5.5.7 affects the administrative controls section of the R.E. Ginna TSs.

- 10 CFR 50.55a(f)(4), requires, in part, that ASME Code Class 1, 2, and 3 components must meet the IST requirements of the ASME OM Code.

- 10 CFR 50.55a(f)(4)(ii) requires, in part, that IST programs be revised every 10 years (120 months) to comply with the requirements of the latest edition and addenda of the ASME OM Code that is incorporated by reference in 10 CFR 50.55a(b).

- 10 CFR 50.55a(f)(5)(ii) requires that if a revised IST program for a facility conflicts with the TS for that facility, the licensee shall apply to the NRC for amendment of the TSs to conform the TS to the revised program. The licensee shall submit this application, as specified in 10 CFR 50.4, at least 6 months before the start of the period during which the provisions become applicable, as determined by 10 CFR 50.55a(f)(4).

- NUREG-1482, Revision 1, "Guidelines for Inservice Testing at Nuclear Power Plants," provides guidelines and recommendations for developing and implementing programs for the IST of pumps and valves at commercial nuclear power plants, the applicable regulations, and the components to be included in an IST program.

- By letter dated December 6, 2005 (ADAMS Accession No. ML053460302), the NRC approved Revision 0 of TSTF-479, "Changes to Reflect Revisions of 10 CFR 50.55a." TSTF-479, Revision 0, proposed to revise references in the Standard Technical Specifications (STS) Administrative Controls IST Program and STS Bases to reflect the current edition of the ASME Code specified in 10 CFR 50.55a(b). The NRC concluded that the proposed revision was acceptable because the requirements of 10 CFR 50.55a adequately provide for IST.

- By letter dated October 4, 2006 (ADAMS Accession No. ML062780321), the NRC approved Revision 0 of TSTF-497, "Limit Inservice Testing Program SR 3.0.2 Application to Frequencies of 2 Years or Less." TSTF-497, Revision 0, proposed to revise the STS IST program by clarifying that the application of the 25-percent IST interval extension allowed by SR 3.0.2 was for IST frequencies of 2 years or less. The NRC concluded that the proposed revision was acceptable because it was an administrative change that clarified that the provisions of SR 3.0.2 (i.e., the 25-percent interval extension) are applicable to IST intervals of 2 years or less.

3.0 TECHNICAL EVALUATION

3.1 Requested Changes

The licensee requested the following changes be made to R.E. Ginna TS 5.5.7: (1) replace references to Section XI of the ASME Code with references to the ASME OM Code, and (2) revise TS 5.5.7.b to state, "The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities."

3.2 Basis for Requested Changes

The proposed changes to the ASME Code references would revise R.E. Ginna TS 5.5.7 in accordance with the requirements of 10 CFR 50.55a(f)(4) and 10 CFR 50.55a(f)(5)(ii).

Revising TS 5.5.7.b to state, "The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the

Inservice Testing Program for performing inservice testing activities," clarifies that the 25-percent interval extension provision of SR 3.0.2 is only applicable to IST intervals of 2 years or less.

3.3 Evaluation of Requested Changes

R.E. Ginna TS 5.5.7 establishes the SRs for IST of ASME Code Class 1, 2, and 3 components, including applicable supports. TS 5.5.7 currently references Section XI of the ASME Code for establishing the IST frequency requirements. On September 11, 2009, the licensee submitted a letter to the NRC regarding its fifth 10-year IST program for safety-related pumps and valves (ADAMS Accession No. ML092610435). This letter established and defined the pump and valve IST program for R.E. Ginna's fifth 10-year interval from January 1, 2010, through December 31, 2019. The licensee stated that the program had been developed as required by 10 CFR 50.55a(f), in accordance with the 2004 Edition of the ASME OM Code. As a result of the licensee updating its IST program to comply with the 2004 Edition of the ASME OM Code pursuant to 10 CFR 50.55a(f)(4)(ii), the reference in R.E. Ginna TS 5.5.7 to Section XI of the ASME Code is no longer applicable. This amendment revises the R.E. Ginna TSs to reference the current ASME OM Code requirements to be used in the fifth 10-year IST program interval beginning on January 1, 2010.

The TS change does not eliminate any tests or relieve the licensee of its responsibility to seek relief from Code test requirements when required. The proposed change of the ASME Code reference from "ASME Section XI" to "ASME OM Code" eliminates the ASME Code inconsistency between the IST program and TS 5.5.7, as required by 10 CFR 50.55a(f)(5)(ii); therefore, the NRC staff finds these proposed changes to be acceptable. This change is also consistent with the NRC's basis for approval of TSTF-479 because the requirements of 10 CFR 50.55a adequately provide for IST.

Revising TS 5.5.7.b to state, "The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities," would clarify that the 25-percent interval extension provision of SR 3.0.2 is only applicable to IST intervals of 2 years or less. This change recognizes the IST program may direct additional tests be performed in accordance with the ASME OM Code at intervals other than those listed in TS 5.5.7. Applying the 25-percent extension permitted by SR 3.0.2 to frequencies in excess of 2 years, such as 5 or 10 years, as permitted by the ASME OM Code, in certain cases, may be inappropriate; therefore, the NRC approved the TSTF-497 proposal to revise STS 5.5.7.b to allow the 25-percent extension only for surveillance intervals of 2 years or less. The NRC approved Revision 0 of TSTF-497 because it was an administrative change that clarified that the provisions of SR 3.0.2 (i.e., the 25-percent interval extension) are applicable to IST intervals of 2 years or less.

The 25-percent extension provides operational flexibility but does not significantly degrade the reliability that results from performing the surveillance at the specified frequency. The proposed change is consistent with guidance contained in Revision 1 of NUREG-1482 regarding maximum allowable extensions of test intervals. This amendment is consistent with the NRC's basis for approving TSTF-497 because it is an administrative change that clarifies that the provisions of SR 3.0.2 (i.e., the 25-percent interval extension) are only applicable to IST intervals of 2 years or less. This amendment is also consistent with the NRC staff position contained in Revision 1 of NUREG-1482; therefore, the amendment is acceptable.

Based on the above evaluation, the NRC staff finds the proposed revisions to the R.E. Ginna TSs meet the requirements of 10 CFR 50.55a and 10 CFR 50.36. Therefore, the staff finds the proposed changes acceptable.

4.0 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.

5.0 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 (74 FR 56887). 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.

6.0 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.

Principal Contributor: Audrey Klett, NRR

Date: April 8, 2010

April 8, 2010

Mr. John T. Carlin
Vice President R. E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT - AMENDMENT RE: REVISIONS TO INSERVICE TEST PROGRAM TECHNICAL SPECIFICATIONS (TAC NO. ME2269)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 110 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 18, 2009.

The amendment revises TS 5.5.7, "Inservice Testing Program," by incorporating TS Task Force Traveler (TSTF) 479, "Changes to Reflect Revision of 10 CFR [Title 10 of the *Code of Federal Regulations*] 50.55a," and TSTF-497, "Limit Inservice Testing Program SR [Surveillance Requirement] 3.0.2 Application to Frequencies of 2 Years or Less." Specifically, the amendment (1) replaces references to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, with the ASME Code for Operation and Maintenance of Nuclear Power Plants for inservice testing activities, and (2) applies the extension allowance of SR 3.0.2 to other normal and accelerated inservice testing frequencies of 2 years or less that were not included in the frequencies listed in TS 5.5.7.a.

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

Sincerely,
/RA/
Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures:
1. Amendment No. 110 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: Distribution via Listserv
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ADAMS Accession No.: ML100740107

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DATE	03 /29/ 10	03 / 26 / 10	02 / 22 / 10	03 /31/ 10	04 /05/ 10	04 /08/ 10

Official Record Copy

DATED: April 8, 2010

AMENDMENT NO. 110 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

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