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

August 12, 2009

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: CONTAINMENT OPERABILITY DURING REFUELING OPERATIONS (TAC NO. ME0203)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 107 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated December 4, 2008.

The amendment revises the Technical Specifications to allow refueling operations with both containment personnel interlock doors to be open under administrative control consistent with Technical Specification Task Force (TSTF) Travelers TSTF-68 and TSTF-312. In support of this amendment request, the licensee recalculated the fuel gas gap fractions for its design basis fuel handling accident and has justified a shorter decay time of 72 hours utilizing the alternative source term methodology.

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. 107 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. 107
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 December 4, 2008, 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. 107, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

CFLyon for

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: August 12, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 107

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
3.9.3-1	3.9.3-1
3.9.3-2	3.9.3-2

(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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 107, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Fire Protection

(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

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch shall be either:

1. bolted in place with at least one access door closed or capable of being closed under administrative control,

2. isolated by a closure plate that restricts air flow from containment with the associated emergency egress door closed or capable of being closed under administrative control, or

3. isolated by a roll up door and enclosure building with the roll up door closed or capable of being closed under administrative control.

b. One door in the personnel air lock shall be closed or capable of being closed under administrative control; and

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

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS. <u>AND</u>	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 107 TO 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 dated December 4, 2008 (Agencywide Document Access and Management System (ADAMS) Accession No. ML083460051), the R.E. Ginna Nuclear Power Plant, LLC (the licensee) submitted a request for changes to the R.E. Ginna Nuclear Power Plant Technical Specifications (TSs). The requested changes would revise the TSs to permit refueling operations with both containment personnel interlock doors maintained open under administrative controls consistent with Technical Specification Task Force (TSTF) Travelers TSTF-68, "Containment Personnel Airlock Doors Open During Fuel Movement," and TSTF-312, "Administratively Control Containment Penetrations." In support of this amendment request, the licensee recalculated the fuel gas gap fractions for its design basis fuel handling accident (FHA) and has justified a shorter decay time of 72 hours utilizing the alternative source term (AST) methodology.

The Nuclear Regulatory Commission (NRC) staff has reviewed the dose analysis for the design basis FHA for Ginna, as well as the meteorological data (i.e., χ /Q values) that served as input to the dose analysis using the AST methodology.

2.0 REGULATORY EVALUATION

A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67, "Accident source term," to voluntarily revise their current accident source term used in design basis radiological consequence analyses. A revised accident source term will be acceptable if the licensee can make the following demonstration:

Paragraph (b)(2)(i) of 10 CFR 50.67 states, "An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE)." Paragraph (b)(2)(ii) of 10 CFR 50.67 states, "An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE)." In addition Paragraph (b)(2)(iii) of 10 CFR 50.67 restates the applicability of the control room habitability criteria of GDC-19 for use of an AST, "Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident."

A licensee who seeks to revise its current accident source term in design basis radiological consequence analyses shall apply for a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." The application shall contain an evaluation of the consequences of applicable design-basis accidents previously analyzed in the safety analysis report.

The regulatory requirements for which the NRC staff based its acceptance are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and 10 CFR Part 50 Appendix A, General Design Criterion 19 (GDC-19), "Control room." The NRC staff also utilized the regulatory guidance provided in the following documents, except where the licensee proposed an acceptable alternative:

- Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms"
- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors"
- RG 1.194, "Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments at Nuclear Power Plants"
- SRP Section 2.3.4, "Short-Term Diffusion Estimates for Accidental Atmospheric Releases"

The NRC staff also considered relevant information in the Ginna Updated Final Safety Analysis Report, TSs, and the licensee's letters dated June 30, 2004 (ADAMS Accession No. ML041890281), August 31, 2005 (ADAMS Accession No. ML052510419) and April 6, 2007 (ADAMS Accession No. ML071010544), which provided additional information on the licensee's August 4, 2003 (ADAMS Accession No. ML032230027), response to Generic Letter (GL) 2003-01, "Control Room Habitability." Additionally, the staff performed independent confirmatory calculations to evaluate the licensee's revised dose analyses. The computer code discussed in NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation," and its supplements, were used by the staff in performing independent dose calculations.

3.0 TECHNICAL EVALUATION

Access to the Ginna containment can be provided through the containment personnel hatch (i.e., personnel air lock). The personnel hatch consists of a hatch barrel with a leak-tight personnel access door on each end that are interlocked in order to prevent simultaneous opening. Currently, TS 3.9.3, "Containment Penetrations," requires that one door in the personnel air lock shall be closed during refueling operations. The licensee has requested a change to TS 3.9.3 to allow refueling operations with both personnel interlock doors to be open provided that one door is capable of being closed under administrative controls.

Ginna Amendment No. 98, which was consistent with TSTF-68 and TSTF-312, was approved on May 5, 2006 (ADAMS Accession No. ML061600223). The amendment revised TS 3.9.3, "Containment Penetrations," to allow an emergency egress door, access door, or roll up door, as associated with the containment equipment hatch penetration, to be closed or capable of being closed during core alterations or movement of irradiated fuel within containment. In the proposed amendment application, the licensee has proposed additional containment/auxiliary building doors to be open during fuel handling operations.

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the impacts of the proposed application of an AST for the Ginna FHA. The staff performed independent calculations to confirm the conservatism of the licensee's analyses. The findings of this safety evaluation (SE) input are based on the descriptions of the licensee's analyses and other supporting information docketed by the licensee.

3.1 Accident Source Term

A full implementation of the AST in accordance with 10 CFR 50.67 was approved as part of Amendment No. 87 to the Ginna TSs on February 25, 2005 (ADAMS Accession No. ML050320491). The licensee used the assumptions in RG 1.183, Table 3 for the basis of the non-loss-of-coolant accident (non-LOCA) fraction of fission product inventory in the gap for the Ginna FHA analysis. The licensee used this accident source term as the bounding fission product inventory.

Footnote 11 to Table 3 of RG 1.183 provides peak power and burnup criteria for use in determining gap fractions. Footnote 11 also includes provisions to provide a plant-specific analysis if these criteria are exceeded. The licensee determined that a small number of fuel rods in future core reloads could exceed the footnote criteria. Therefore, as part of this proposed license amendment submittal, the licensee provided revised plant specific conservative values for non-LOCA gap fractions to be used in the revised design basis FHA.

As an alternative to RG 1.183 Table 3, the licensee used a method to determine its release fractions based on ANSI/ANS-5.4-1982, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel." This standard provides an analytical method for calculating the release of volatile fission products from oxide fuel pellets during normal reactor operation. When used with nuclide yields, this method will give the so-called "gap activity," which is the inventory of volatile fission products that could be available for release from the fuel rod if the cladding were breached.

ANSI/ANS-5.4-1982 considers high-temperature (up to the melting point) and low-temperature (where temperature-independent processes dominate) releases and distinguishes between short-half life (half life less than 1 year) and long-half life (half life greater than 1 year) nuclides. This standard requires that releases for nuclides of interest be calculated with both the high-temperature and the low-temperature models, and the larger of the two calculated releases is to be taken as the result.

In its approach, the licensee used beginning-of-life (BOL), end-of-life (EOL), and peak thermal power axial power profiles combined with the peak pin power versus burnup and an instrument error uncertainty factor of 1.02 to yield limiting projected power histories for Ginna. The fuel temperature radial and axial profiles were then modeled using the Studsvik computer code INTERPIN, which conservatively showed large increases in fuel temperature with increasing burnup. The worst-case release fraction for each isotope was extracted from these cases consistent with the ANSI/ANS-5.4-1982 methods, and a bounding value was applied to bound all cases. For additional conservatism, the licensee used bounding gas gap release fractions applied to all failed fuel rods, regardless of burnup or power level.

The licensee's results indicate that gas gap fractions must be significantly increased for the limiting pins with burnups over 54 giga-watt days (GWd) per metric ton uranium (MTU) and with linear heat generation rates in excess of 6.3 kw/ft. The licensee determined that doubling of the I-131, I-132, I-134, I-135, Xe-135, Xe-135m, Xe-138, Kr-85, Kr-85m, Kr-87, and Kr-88 gas gap release fractions and tripling of the I-133, Xe-133, and Xe-133m gas gap release fractions yield bounding and conservative results. All of the gas gap activity in the damaged rods is assumed to be released and consists of 20% of the Kr-85, 15% of Xe-133 and Xe-133m, 10% of the other noble gases, 16% of the I-131, 15% of I-133, and 10% of the other iodine isotopic inventories at the time of the accident.

To account for gap fraction uncertainty in fuel that does not meet the criteria specified in footnote 11 of RG 1.183, the licensee multiplied these gap fractions by at least a factor of two and in some cases tripled the gap release fractions based on results of analysis. This conservative approach is acceptable to the NRC staff and has been previously approved by the NRC as cited by the licensee.

The NRC staff finds that the licensee's assumptions for the source term and power level are appropriately conservative and are based on a methodology that provides an acceptably conservative alternative to RG 1.183 Table 3 and is, therefore, acceptable for use in the Ginna non-LOCA AST dose consequence analysis events.

3.2 Fuel Handling Accident Radiological Consequence Analysis

3.2.1 Analysis Summary

A design-basis FHA during refueling could release some fraction of the fission product inventory to the environment. Two accident scenarios are considered in the Ginna analysis:

- A refueling accident in containment
- A refueling accident in the auxiliary building.

The possibility of a FHA incident is very remote because of the administrative controls and physical limitations imposed on fuel handling operations. These refueling limitations and other fuel design specifications would likely preclude fuel cladding integrity failures during any fuel handling operations including the unlikely events evaluated by the licensee for Ginna. The two design-basis accident (DBA) scenarios using the AST for Ginna are evaluated to determine a bounding dose consequence for a fuel handling DBA, referred to as an AST FHA, and to evaluate the adequacy of the licensee credited engineered safety features.

The proposed Ginna AST FHA analysis evaluated the dose consequence following movement of fuel that has decayed a minimum of 72 hours since it occupied part of a critical reactor core.

The FHA in the Ginna containment or the auxiliary building begins with the dropping of a fuel assembly, which conservatively assumes that all of the fuel rods in the assembly are damaged. Volatile constituents of the core fission product inventory are assumed to migrate from the fuel pellets to the gap between the pellets and the fuel rod clad. It is then assumed that the entire gap inventory is instantaneously released into either the containment reactor cavity or auxiliary building fuel pool water. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water in the reactor cavity or spent fuel pool. Partitioning of the iodine occurs in the water before entering the atmosphere; noble gases escape without benefit of partitioning. The guidance in RG 1.183 allows an effective halogen decontamination factor of 200 when the overlaying water column is at least 23 feet. In addition, the licensee assumed retention of all aerosol and particulate radionuclide's within the spent fuel pool water as allowed by RG 1.183 Appendix B, Regulatory Position 3.

The licensee determined that since 1) the auxiliary building vent pathway is filtered and 2) the containment equipment hatch pathway will result in a direct ground level release that bound other potential release points in the intermediate building and auxiliary building, the equipment hatch pathway results in the more conservative off-site and control room doses for this analyzed AST FHA. This pathway was also determined to be limiting in the licensee's previous AST FHA analysis approved in Ginna Amendment No. 98.

3.2.2 Fuel Handling Accident Transport

The licensee assumed that 100% of the radionuclides released from the reactor cavity in containment are released to the environment in 2 hours without any credit for filtration, holdup, or dilution. For its analyzed FHA in the auxiliary building spent fuel pool, the licensee assumed iodine removal by the auxiliary building ventilation system charcoal filters (90% for elemental iodine and 70% for organic iodine). These radioactive transport assumptions are consistent with the guidance provided in RG 1.183.

3.2.3 Control Room (CR) Habitability for the Fuel Handling Accident

The Ginna CR Emergency Air Treatment System (CREATS) is normally in standby and is configured to provide zone isolation, re-circulation and filtration under accident conditions. The system is designed to provide a protective environment from which the operators can control the plant for 30 days after a DBA without exceeding associated dose reference limits. The CREATS is also designed to protect the operators from exposure to toxic gas following an accidental release from sources on or near the Ginna site. The CREATS consists of two seismic category 1, 100% capacity trains that are designed to filter, cool, heat, and recirculate 6000 cfm (± 10%) of control room air. The CREATS fans are powered from Class 1E safeguard buses and will start upon a manual, toxic gas, radiation, or safety injection (SI) signal.

The licensee assumed a control room isolation delay of 60 seconds to account for damper positioning and instrumentation delays and an additional 10 seconds for the CREATS to be operational following an AST FHA. Following isolation, there will be no outside air makeup and the filtered recirculation flow of 5400 cfm (6000 cfm nominal less 10%) is initiated. The licensee assumed an unfiltered inleakage rate of 300 cfm and recirculation filter efficiencies of 90%, 70%, and 98% for elemental, organic, and particulate iodine, respectively.

3.2.4 FHA Conclusion

The NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the radiological impacts of the proposed license amendment changes. The staff reviewed the Ginna AST FHA radiological consequence analysis information provided in the licensee's submittal and performed a confirmatory calculation. The staff finds that the licensee used analysis, methods, and assumptions that are consistent with the guidance of RG 1.183. The staff also finds that the licensee employed conservative assumptions with regards to operability of potential dose mitigation systems. The licensee's calculated dose results are provided in Table 1 of this SE. The licensee's assumptions, which have been found acceptable to the staff, are listed in Tables 2 and 3 of this SE.

The NRC staff compared the AST FHA doses estimated by the licensee to the applicable acceptance criteria. The staff has found the licensee's calculated doses acceptable because they are within the 10 CFR 50.67 requirements, RG 1.183 guidelines, and the SRP 15.0.1 radiological dose acceptance criteria for a FHA using an AST methodology. These criteria are 6.3 roentgen equivalent man (rem) TEDE at the exclusion area boundary (EAB) for the worst 2 hours, 6.3 rem TEDE at the low population zone (LPZ) for the duration of the accident, and 5 rem TEDE in the CR for the duration of the accident.

3.3 Atmospheric Dispersion

The licensee used 5 years of onsite hourly meteorological data collected during calendar years 1999 through 2003 to identify the limiting CR atmospheric dispersion factor (χ /Q value) for input into the FHA dose assessment which assumes refueling operations with both personnel airlock doors open under administrative control. These meteorological data have been utilized in prior license amendment requests (LARs) and are discussed in the SE associated with Ginna Amendment No. 87.

The licensee generated new CR χ /Q values for postulated ground level releases for the current LAR using guidance provided in RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants." These new χ /Q estimates were calculated using the ARCON96 atmospheric dispersion computer code NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes". RG 1.194 states that ARCON96 is an acceptable methodology for assessing onsite χ /Q values for use in DBA radiological analyses. The NRC staff evaluated the applicability of the ARCON96 model and concluded that there are no unusual sitting, building arrangements, release characterizations, source-receptor configurations, meteorological regimes, or terrain conditions that preclude use of this model in support of the current LAR for Ginna.

The licensee assumed that FHA releases from the containment building through the environment to the CR air intake would occur by the most limiting pathway associated with an isolated containment; an open equipment hatch (EH), an open personnel air lock (PAL) or an open or operating containment vent. The licensee stated that the analysis supports an open PAL during fuel movement in containment, an open EH during fuel movement in containment, and an open roll-up door in the south wall of the auxiliary building during fuel movement in the spent fuel pool. The licensee generated a number of χ /Q values and determined that a postulated point source release from the EH roll-up door to the CR air intake would be the limiting case. The NRC staff performed a qualitative comparison of the licensee's calculations.

The licensee used previously generated χ /Q values to assess the radiological consequences of the postulated FHA for the EAB and LPZ. These X/Q values are discussed in the SE associated with Ginna Amendment No. 87.

The NRC staff has previously reviewed and approved a meteorological analysis for Ginna associated with Ginna Amendment No. 87, Amendment No. 97 dated February 15, 2006 (ADAMS Accession No. ML060950487), and Amendment No. 98, and found them consistent with the inputs to the ARCON96 calculations and staff practice. As part of the current LAR, the licensee generated additional CR χ /Q values. The staff did not review these values other than to qualitatively confirm that the licensee had correctly identified the limiting CR χ /Q value which is more limiting than the comparable current licensing basis χ /Q value. The staff finds use of the limiting CR χ /Q value of 6.90×10^{-3} sec/m^3 acceptable in support of the current LAR, to allow refueling operations with one personnel air lock door closed or capable of being closed under administrative control. Based upon the review described in the SE associated with Ginna Amendment No. 87, the staff concluded that the Ginna EAB and LPZ χ /Q values were acceptable. With regard to the current LAR, the staff's acceptance of χ /Q values is limited to the χ /Q values which appear in Table 2 of this SE.

3.4 Technical Specification Changes

TS 3.9.3, Limiting Condition for Operation (LCO) 3.9.3.b currently requires that one door in the personnel air lock be closed during refueling operations. To meet this requirement, it is necessary to have the interlocks in-service and to cycle the air lock doors frequently during refueling operations. The licensee states that this requirement involves excessive cycling of these access doors which results in damage and costly repairs. In addition, the licensee states, "The door interlocks could delay the evacuation of personnel in the event of an emergency inside containment." To alleviate this potential safety hazard and unnecessary expense, the licensee has requested a revision to TS LCO 3.9.3.b to require that the personnel air lock have one door closed or capable of being closed under administrative control. This will allow the flexibility to perform refueling operations with both airlock doors open. This is consistent with the intentions of NUREG 1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.1, TSTF-68 and TSTF-312.

The proposed change to TS 3.9.3 creates a potential release path through the personnel air lock doors for a FHA within containment. The credible release points to the environment from the personnel air lock doors would be either the plant vent and/or the limiting release points from the intermediate building structure. As discussed above, the licensee has determined that the previously evaluated equipment hatch for the AST as implemented in Amendment No. 87 is the bounding release point for the Ginna AST FHA.

The NRC staff has previously reviewed and approved TSTF-312 as well as TSTF-68 and the licensee has committed to implement administrative requirements to close one personnel hatch door within 30 minutes of a FHA within containment. The NRC staff finds, with reasonable assurance, that Ginna will continue to provide sufficient safety margins with adequate defense in depth in the unlikely event of a design-basis FHA event. Therefore, the staff concludes that the proposed AST FHA implementation is acceptable from the standpoint of radiological consequences. Therefore, the staff finds the licensee's proposed TS change acceptable.

Table 1

Ginna AST FHA Radiological Consequences Expressed as TEDE (1) (rem)

Design Basis Accident	EAB (2)	LPZ (3)	CR (4)
Fuel Handling Accident (Containment Cavity)	1.48E+00	1.71E-01	4.04E+00
Fuel Handling Accident (Spent Fuel Pool)	1.48E+00	1.71E-01	8.08E-01
Dose Criteria	6.30E+00	6.30E+00	5.00E+00

(1) Total effective dose equivalent
(2) Exclusion area boundary
(3) Low population zone
(4) Control Room

Note: Licensee results are expressed to a limit of two significant figures

Table 2

Ginna AST FHA Containment Equipment Hatch Open

χ/Q Values (sec/m^3)

Time Interval	Receptor	χ/Q Value (s/m^3)
0 - 2 hr	Control room	6.90×10^{-3}
0 - 2 hr	EAB	2.17×10^{-4}
0 - 8 hr	LPZ	2.51×10^{-5}

Table 3
Ginna AST Data and Assumptions for the FHA

Power level	1775 MWt
Power factor	1.02
Pin peaking factor	2.6
Fuel assembly damaged	1
Fuel pool water depth	23
Fission product decay period	72 hours
Duration of accident	2 hours
Containment personnel air lock	Open
Containment equipment Hatch	Open
South wall auxillary building roll-up door	Open
Release filtration or holdup	None credited
AST FHA release point	Containment Equipment Hatch
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
CREATS initiation	70 seconds after FHA
CREATS filtered recirculation flow	5400
CREATS unfiltered inleakage	300 cfm

Ginna Gas Gap Fractions		
Isotope	RG 1.183	Ginna
I-131	0.08	0.16
I-132	0.05	0.10
I-133	0.05	0.15
I-134	0.05	0.10
I-135	0.05	0.10
Xe-133	0.05	0.15
Xe-133m	0.05	0.15
Xe-135	0.05	0.10
Xe-135m	0.05	0.10
Xe-138	0.05	0.10
Kr-85	0.10	0.20
Kr-85m	0.05	0.10
Kr-87	0.05	0.10
Kr-88	0.05	0.10

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 10311). 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 Contributors: J. Shea, L. Brown

Date: August 12, 2009

August 12, 2009

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: CONTAINMENT OPERABILITY DURING REFUELING OPERATIONS (TAC NO. ME0203)

Dear Mr. Carlin:

The Commission has issued the enclosed Amendment No. 107 to Renewed Facility Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to your application dated December 4, 2008.

The amendment revises the Technical Specifications to allow refueling operations with both containment personnel interlock doors to be open under administrative control consistent with Technical Specification Task Force (TSTF) Travelers TSTF-68 and TSTF-312. In support of this amendment request, the licensee recalculated the fuel gas gap fractions for its design basis fuel handling accident and has justified a shorter decay time of 72 hours utilizing the alternative source term methodology.

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. 107 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: Distribution via Listserv
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OFFICE	LPL1-1\PM	LPL1-1\LA	OGC	AADB\BC	LPL1-1\BC
NAME	DPickett	SLittle	MSmith	RTaylor by memo dated	NSalgado
DATE	08 /03/ 09	08 /03/ 09	08 /07/ 09	07 / 13 / 09	08 /11/ 09

Official Record Copy

DATED: August 12, 2009

AMENDMENT NO. 107 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

PUBLIC	
LPLI-1 R/F	
N. Salgado	RidsNrrDorlLPL1-1
S. Little	RidsNrrLASLittle
D. Pickett	RidsNrrPMDPickett
OGC	RidsOgcMailCenter
ACRS	RidsAcrsAcnwMailCenter
R. Taylor	RidsNrrDraAadb
JShea, AADB	
LBrown, AADB	
G. Dentel, RI	

cc: Plant Service list