February 15, 2006

Mrs. Mary G. Korsnick
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: RELAXED AXIAL OFFSET CONTROL (TAC NO. MC6867)

Dear Mrs. Korsnick:

The Commission has issued the enclosed Amendment No. 94 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 April 29, 2005, as supplemented on September 19, 2005.

The amendment revises the Technical Specifications to incorporate the relaxed axial offset control and heat flux hot channel (F_Q) surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control requirements and to increase the ability to return to power after a plant trip or transient. The changes are consistent with Westinghouse Electric Company Report WCAP-10216-P-A, “Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification.”

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/

Patrick D. Milano, Sr. 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. 94 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML060190605

OFFICE	LPLI-1\PM	LPLI-1\LA	SPWB\BC	OGC	LPLI-1\BC
NAME	PMilano	SLittle	JNakoski		RLaufer
DATE	01/26/06	01/26/06	01/27/06	02/09/06	02/14/06

Official Record Copy

DATED: February 15, 2006

AMENDMENT NO. 94 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R.E. GINNA NUCLEAR POWER PLANT

PUBLIC	
LPLI-1 R/F	
R. Laufer	RidsNrrDorlLpla
S. Little	RidsNrrLASLittle
P. Milano	RidsNrrPMPMilano
J. Nakoski	RidsNrrDssSpwb
F. Akstulewicz	RidsNrrDssSpnb
T. Boyce	RidsNrrDirsItsb
K. Wood	
G. Hill (2)	
OGC	RidsOgcMailCenter
ACRS	RidsAcrsAcnwMailCenter
J. Trapp, RI	

cc: Plant Service list

cc:

Mr. Michael J. Wallace
President
R.E. Ginna Nuclear Power Plant, LLC
c/o Constellation Energy
750 East Pratt Street
Baltimore, MD 21202

Mr. John M. Heffley
Senior Vice President and
Chief Nuclear Officer
Constellation Generation Group
1997 Annapolis Exchange Parkway
Suite 500
Annapolis, MD 21401

Kenneth Kolaczyk, Sr. Resident Inspector
R.E. Ginna Nuclear Power Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, NY 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Peter R. Smith, President
New York State Energy, Research,
and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Carey W. Fleming, Esquire
Senior Counsel - Nuclear Generation
Constellation Generation Group, LLC
750 East Pratt Street, 17th Floor
Baltimore, MD 21202

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Ms. Thelma Wideman, Director
Wayne County Emergency Management
Office
Wayne County Emergency Operations
Center
7336 Route 31
Lyons, NY 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
1190 Scottsville Road, Suite 200
Rochester, NY 14624

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

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. 94
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 April 29, 2005, as supplemented on September 19, 2005, 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. 94, 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 the date of its issuance and shall be implemented prior to startup from the fall 2006 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 15, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 94

RENEWED FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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
3.2.1-1	3.2.1-1
3.2.1-2	3.2.1-2
3.2.1-3	3.2.1-3
3.2.1-4	3.2.1-4
3.2.3-1	3.2.3-1
3.2.4-1	3.2.4-1
3.2.4-2	3.2.4-2
3.2.4-3	3.2.4-3
3.3.1-1	3.3.1-1
3.3.1-2	3.3.1-2
3.3.1-3	3.3.1-3
3.3.1-4	3.3.1-4
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.3.1-7	3.3.1-7
3.3.1-8	3.3.1-8
3.3.1-9	3.3.1-9
3.3.1-10	3.3.1-10
3.3.1-11	3.3.1-11
3.3.1-12	3.3.1-12
3.3.1-13	3.3.1-13
3.3.1-14	3.3.1-14
3.3.1-15	3.3.1-15
3.3.1-16	3.3.1-16
5.6-1	5.6-1
5.6-2	5.6-2
5.6-3	5.6-3
5.6-4	5.6-4
5.6-5	5.6-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 94 TO RENEWED FACILITY

OPERATING LICENSE NO. DPR-18

R.E. GINNA NUCLEAR POWER PLANT, INC.

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated April 29, 2005, as supplemented on September 19, 2005 (References 1 and 2; Agencywide Documents Access and Management System Accession Nos. ML051300330 and ML052720369), R.E. Ginna Nuclear Power Plant, LLC. (Ginna LLC, or the licensee) submitted a request for changes to the R.E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TSs). The requested changes would incorporate relaxed axial offset control (RAOC) and heat flux hot channel (F_Q) surveillance methodologies. These methodologies are used to reduce operator action required to maintain conformance with power distribution control requirements and to increase the ability to return to power after a plant trip or transient. The RAOC methodology allows the axial flux difference control bands to be widened by removing conservatism inherent in the constant axial offset control (CAOC) methodology currently used at Ginna. The proposed changes are in accordance with Westinghouse Electric Company Report WCAP-10216-P-A, “Relaxation of Constant Axial Offset Control/F_Q Surveillance Technical Specification” (Reference 3). The requested changes would also move requirements for the axial flux difference (AFD) and quadrant power tilt ratio (QPTR) monitor alarms to the Technical Requirements Manual (TRM). The September 19, 2005, letter provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 7, 2005 (70 FR 33220)

2.0 REGULATORY EVALUATION

2.1 Power Distribution Limits

During power operation, the global power distribution is monitored by the AFD and the quadrant power tilt ratio QPTR, which are directly and continuously measured process variables. Local power peaking is monitored by $F_Q(z)$, and nuclear enthalpy rise hot-channel factor, ($F^N_{\Delta H}$). Limits on these parameters ensures that the nuclear core limits are not exceeded.

AFD is a measure of the balance of power between the top and bottom halves of the core. Limits are established on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

QPTR is a measure of the balance of power between the radial quadrants of the core. Limits are established on the values of the QPTR in order to limit the radial power distribution skewing to a quadrant. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting the QPTR prevents radial xenon oscillations and will indicate any core asymmetries.

$F_Q(z)$ is a measure of the local power. The value of $F_Q(z)$ varies along the axial height of the core (z). $F_Q(z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions adjusted for uncertainty. Therefore, $F_Q(z)$ is a measure of the peak pellet power within the reactor core.

$F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. $F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power.

2.2 Current CAOC Methodology

The current TSs stipulate the use of the CAOC methodology for analyzing and controlling the AFD parameter. In the power range, CAOC methodology calls for a constant band about a reference AFD, without regard to reactor power. In the power range, RAOC methodology generates a power dependent AFD curve. Generally, the RAOC allowed AFD increases with decreasing power. In some plants, the CAOC methodology results in significant margin between the CAOC generated AFD and the maximum permitted AFD in the loss-of-coolant accident (LOCA) analysis. RAOC methodology allows a licensee to transfer some of that margin to operating flexibility.

2.3 Regulatory Requirements

The Nuclear Regulatory Commission (NRC) staff based its acceptance on the following regulatory requirements:

Appendix A, “General Design Criteria (GDC) for Nuclear Power Plants,” to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. In particular, GDC 10, “Reactor Design,” states that “[t]he reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.” As noted in the Ginna Updated Final Safety Analysis Report (UFSAR), the design criteria used at Ginna predate Appendix A to 10 CFR Part 50. The Ginna design criteria comprise the proposed Atomic Industrial Forum (AIF) versions of the criteria issued for comment by the Atomic Energy Commission on July 10, 1967. These criteria define or describe safety objectives and approaches incorporated in the design of Ginna.

In February 1978, the NRC initiated its Systematic Evaluation Program (SEP) for eleven operating plants, including Ginna, that had received construction permits between 1956 and 1967. The SEP consisted of a limited review of the designs of these older plants. The purpose of the SEP was to reconfirm and document the design safety because the safety criteria had changed since the plants were originally licensed. As part of the SEP, the original codes and standards used in the design of structures, systems, and components at Ginna were compared with later licensing criteria. The results for Ginna were documented in NRC Report NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R.E. Ginna Nuclear Power Plant, Final Report," December 1982 (ADAMS No. 8309200476). The current UFSAR incorporates the SEP review into the current licensing basis at Ginna. The adequacy of the Ginna design relative to the 1972 version of the GDC is described in Section 3.1.2 of the UFSAR. Specifically, the conformance to GDC 10 at Ginna is described in UFSAR Section 3.1.2.2.1. Therefore, the NRC staff reviewed the application and based its acceptance on the design criteria in GDC 10 rather than the criteria in the 1967 version of the AIF criteria.

The NRC approved Westinghouse Report WCAP-10216-P-A, Revision 1, on November 26, 1993, (Reference 4) for referencing in license applications to the extent specified and under the limitations stated in the topical report and NRC evaluation. Since the NRC staff found the report acceptable, the purpose of its review of the licensee's April 29, 2005, application was to ensure that WCAP-10216-P-A was applicable to Ginna and that WCAP-10216-P-A was being implemented appropriately.

The regulatory requirements related to the content of the TSs are set forth in Section 50.36, "Technical specifications," of 10 CFR Part 50. This regulation requires that the TSs include items in five specific categories. These categories are: (1) safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's regulatory and technical analyses in support of its proposed license amendment, which are described in the licensee's April 29, 2005, application, as supplemented on September 19, 2005.

3.1 Proposed TS Changes

a. The licensee proposed to revise TS 3.2.1, "Heat Flux Hot Channel Factor FQ(Z)," TS 3.2.3, "Axial Flux Difference," and TS 5.6.5, "Core Operating Limits Report (COLR)," to adopt the RAOC calculational procedure as described in NUREG-1431, "Standard Westinghouse Technical Specifications Westinghouse Plants."

b. The licensee proposed changes to TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)" to provide the necessary consistency with the changes made to TS 3.2.1 and TS 3.2.3.

c. The licensee proposed changes to TS 3.3.1, "Reactor Trip System (RTS) Instrumentation," to accommodate the change to the RAOC methodologies by revising the form of the f(ΔI) penalty (f(ΔI) is a function of the indicated difference between the top and bottom detectors of the power range neutron flux channels) for overtemperature delta temperature (ΔT) and for overpower ΔT (initially set to 0).

d. The licensee proposed to relocated the requirements for the QPTR alarm monitor from TS 3.2.4 to the TRM.

e. The licensee proposed to relocated the requirements for the AFD alarm monitor from TS 3.2.3 to the TRM.

f. The licensee proposed to revise the listing of analysis methodologies contained in TS 5.6.5, “Core Operating Limits Report (COLR),” to include refences for the RAOC methodology.

3.2 RAOC/F_Q Surveillences

The NRC’s approval of Westinghouse Report WCAP-10216-P-A was limited to pressurized-water reactors designed by Westinghouse. Ginna meets these applicability criteria. The NRC approval did not impose any additional limitations beyond those already present in WCAP-10216-P-A. WCAP-10216-P-A is comprised of two parts. Part A covers the relaxation of constant axial offset control. Part B covers the F_Q (heat flux hot channel factor) TS surveillance requirements.

Part A describes a method for determining an acceptable AFD profile. AFD is a measure of the power differential between the top and bottom halves of the core. If power is shifted too much in either direction it can exacerbate local peaking, resulting in challenges to fuel integrity. The RAOC method generates a range of AFD shapes at different power levels. Those shapes are used in LOCA analysis, loss of flow accident analysis, anticipated operational occurrence analysis. Unacceptable shapes are eliminated by each subsequent step. The remaining AFD shapes constitute a power dependent AFD profile, which is then used to operate the reactor. As noted previously, the RAOC method identifies margin in the analysis and transfers it to operational flexibility. In its application, as supplemented, the licensee demonstrated and affirmed its ability to implement the WCAP-10216-P-A RAOC methodology.

Part B describes an alternate means for monitoring F_Q. The WCAP-10216-P-A method measures F_Q directly under equilibrium conditions. The method then increases the measured F_Q to account for manufacturing tolerance and measurement uncertainties. A further increase is made to account for the effect of normal operating transients. This final F_Q is compared to TS limits to determine whether this parameter is out of specification. By the application, as supplemented, the licensee demonstrated and affirmed its ability to implement the WCAP-10216-P-A RAOC methodology.

In its proposed implementation of the WCAP-10216-P-A, RAOC/F_Q Surveillance, the licensee stated the supporting analysis was performed at extended power uprate (EPU) conditions, and that the analysis is bounded at the current power level. While the use of WCAP-10216-P-A is not predicated on a particular reactor power or plant configuration, changes thereto could alter the results of analysis when using WCAP-10216-P-A. The licensee submitted an application dated July 7, 2005 (Reference 5), to allow a 16.8% increase in its licensed thermal power level (extended power uprate (EPU)). In order to accommodate the EPU, the licensee is making numerous changes to its facility and analyses. Some of those changes would affect the analyses used to support the RAOC/F_Q Surveillance determination. Two such changes are reflected in license amendment applications dated April 29, 2005, regarding revised LOCA methodologies and changes in main feedwater isolation (References 6 and 7). The synergistic combination of these changes is inherent in the analysis that led to the RAOC/F_Q Surveillance determination. The NRC staff had a concern that, absent those synergistic effects, the analysis used to support the licensee's request may not bound current conditions. In its response to a request for additional information dated September 19, 2005 (Reference 2), the licensee provided a discussion of several items affected by the EPU and associated changes that could impact the analysis used to support the WCAP-10216-P-A acceptability determination, and a potential course of action should those changes not take place in concert with WCAP-10216-P-A implementation. The licensee affirmed that the reload methodology in use at Ginna (WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (Reference 8)) would verify the use of the AFD profile and identify any non-bounding conditions. Therefore, the NRC staff has reasonable assurance the WCAP-10216-P-A, RAOC/F_Q Surveillance, methodologies will be implemented appropriately.

The changes to the Ginna TSs being made to incorporate WCAP-10216-P-A, RAOC/F_Q Surveillance, are consistent with NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.0 (Reference 9), and are acceptable.

Based on the above evaluations, the NRC staff has determined that Westinghouse Report WCAP-10216-P-A is applicable to Ginna and that the licensee's request to implement the report is acceptable.

3.3 <u>Relocation of QPTR and AFD Monitor Alarm Requirements</u>

The QPTR is monitored on an automatic basis using the plant process computer that has a QPTR monitor alarm. The plant process computer determines the ratio of the highest average power in any quadrant to the average power in the 4 quadrants by using the inputs from the excore detectors. If the calculated QPTR exceeds the 1.02 limit, an alarm message is generated so that the operators can take corrective action.

The AFD is monitored on an automatic basis using the plant process computer that has an AFD monitor alarm. The computer determines the 1 minute average of each of the operable excore detector outputs and provides an alarm message immediately if the AFD for two or more operable excore channels is outside its specified limits.

Originally, the requirements of 10 CFR 50.36 established the categories of items for inclusion in the TSs, but not the particular requirements for the TSs of each individual plant. The NRC provided guidance for the specific contents of the TSs in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement), 58 FR 39132 (July 22, 1993). In particular, the NRC indicated that certain items could be relocated from the TSs to licensee-controlled documents. The Final Policy Statement established four criteria for determining the items required for inclusion in the TSs:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

These criteria have been codified in 10 CFR 50.36(c)(2)(ii). See Final Rule, "Technical Specifications," 60 FR 36953 (July 19, 1995). As a result, TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TSs. The Final Policy Statement allows the relocation of items not meeting these four specified criteria from the TSs to licensee-controlled documents, such that future changes can be made to these provisions pursuant to 10 CFR 50.59. The NRC also concluded that compliance with the Final Policy Statement satisfied Section 182a of the Act, precipitating a revision to 10 CFR 50.36 which superseded the Final Policy Statement. Thus, within this general framework, licensees may remove material from their TSs on two conditions: (1) the material is not required to be in the TSs based on the NRC staff's interpretation of 10 CFR 50.36, including judgments about the level of detail required in the TSs, and (2) there exist suitable alternative regulatory controls for the material.

The licensee proposed a revision to TS 3.2.3 in order to relocate the portion of the LCO and associated Action and SRs regarding the AFD alarm monitor that are not required to be contained in the TSs. The licensee proposed a revision to TS 3.2.4 in order to relocate the portion of the LCO and associated SRs regarding the QPTR alarm monitor that are not required to be contained in the TSs. These requirements will be relocated to the TRM, which is a licensee-controlled document that is subject to the provisions of 10 CFR 50.59 and provides an appropriate level of review and approval for the revision of requirements that are important to safety, but do not satisfy the criteria of 10 CFR 50.36(c)(2)(ii) for TS requirements. Changes to the TRM requirements are subject to the regulations of 10 CFR 50.59 because the licensee has incorporated the TRM into the Ginna UFSAR. The proposed revision will maintain an appropriate level of control of the relocated requirements and an improved level of consistency with NUREG-1431, which does not contain requirements for the TSs proposed for relocation. The application of these requirements as part of the TRM will be identical to those applicable to the TSs.

The proposed change will relocate the identified TSs to the licensee-controlled TRM consistent with the 10 CFR 50.36 requirements. The NRC staff has reviewed the licensee's submittal and finds that relocation of these requirements to a licensee-controlled document is acceptable in that the LCO and associated requirements were found not to fall within the scope of the criteria contained in 10 CFR 50.36(c)(2)(ii), and changes to licensee-controlled documents will be adequately controlled by 10 CFR 50.59, as applicable.

Relocating the requirements for the AFD and QPTR monitor alarms to the TRM is consistent with TSTF-110, “Delete SR Frequencies Based on Inoperable Alarms,” (Ref. 10).

3.4 Summary

The NRC staff has reviewed the license amendment request and has concluded that the safety margin will remain acceptable and Ginna continues to meet regulatory requirements. Since Ginna is a Westinghouse plant and the RAOC control limits will be determined by the approved Westinghouse LOCA and non-LOCA methods, the NRC staff concludes the licensee’s use of the methods documented in WCAP-10216-P-A is 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 (70 FR 33220). 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.

7.0 REFERENCES

1. Ginna LLC letter, M. G. Korsnick to NRC, “License Amendment Request Regarding Adoption of Relaxed Axial Offset Control (RAOC), R.E. Ginna Nuclear Power Plant, Docket No. 50-334," dated April 29, 2005. (ML051300330)

2. Ginna LLC letter, M. G. Korsnick to NRC, “Response to NRC Generic Request for Additional Information Regarding Relaxed Axial Offset Control, R. E. Ginna Nuclear Power Plant, (TAC No. MC6867)," dated September 19, 2005. (ML052720369)

3. Westinghouse Report WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control/FQ Surveillance Technical Specification," Revision 1, February 1994.

4. NRC letter, Ashok C. Thadani, Director, Division of Systems Safety and Analysis, to Nicholas J. Liparulo, Manager, Nuclear Safety and Regulatory Activities, Westinghouse Electric Corporation, “Acceptance for Referencing of Revised Version of Licensing Topical Report WCAP-10216-P-A, Rev. 1, Relaxation of Constant Axial Offset Control - F_Q Surveillance Technical Specification," (TAC No. M88206), dated November 26, 1993.

5. Ginna LLC letter, M. G. Korsnick to NRC, “R.E. Ginna Nuclear Power Plant, Docket No. 50-244, License Amendment Request Regarding Extended Power Uprate," dated July 7, 2005. (ML051950123)

6. Ginna LLC letter, M. G. Korsnick to NRC, “License Amendment Request Regarding Revised Loss of Coolant Accident (LOCA) Analyses-Changes to Accumulator, Refueling Water Storage (RWST),and Administrative Control Technical Specifications, R.E. Ginna Nuclear Power Plant, Docket No. 50-334," dated April 29, 2005. (ML051260239)

7. Ginna LLC letter, M. G. Korsnick to NRC, “License Amendment Request Regarding Main Feedwater Isolation Valves, R.E. Ginna Nuclear Power Plant, Docket No. 50-334," dated April 29, 2005. (ML051260236)

8. Westinghouse Report WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

9. NRC Report NUREG-1431, “Standard Technical Specifications Westinghouse Plants,” Revision 3.0.

10. TSTF-110, “Delete SR Frequencies Based on Inoperable Alarms,” Revision 2, October 2, 1997. (ML040490071)

Principal Contributor: K. Wood

Date: February 15, 2006