December 8, 2005

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: ONE-TIME EXTENSION OF CONTAINMENT INTEGRATED LEAKAGE RATE TEST INTERVAL (TAC NO. MC6375)

Dear Mrs. Korsnick:

The Commission has issued the enclosed Amendment No. 93 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 March 10, 2005, as supplemented on June 8 and August 31, 2005.

The amendment revises Technical Specification (TS) 5.5.15, "Containment Leakage Rate Testing Program," to extend, on a one-time basis, the interval for completing the next containment integrated leakage rate test, pursuant to Appendix J to Part 50 of Title 10 of the *Code of Federal Regulations*, from 10 years to 15 years since the last test. Therefore, the first test performed after the May 31, 1996, test shall be performed by May 31, 2011.

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, 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. 93 to Renewed License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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Accession Number: ML53130384 Package No.:

OFFICE	NRR\LPLI-1/PM	NRR/LPLI-1\LA	EEMB\BC	ACVB\BC	APLA\BC	OGC	NRR\LPLI-1SC
NAME	PMilano	SLittle	KManoly	RDennig	MRubin		RLaufer
DATE	11/16/05	11/16/05	10/06/05	11/04/05	11/04/05	11/29/05	12/07/05

Official Record Copy

DATED: December 8, 2005

AMENDMENT NO. 93 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18
R. E. GINNA NUCLEAR POWER PLANT

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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. 93
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 March 10, 2005, as supplemented on June 8 and August 31, 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: that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and 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. 93, 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 within 90 days.

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: December 8,2005

ATTACHMENT TO LICENSE AMENDMENT NO. 93

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
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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 93 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 March 10, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050760381), as supplemented on June 8 and August 31, 2005 (ADAMS Nos. ML051660282 and ML052510416, respectively), the R.E. Ginna Nuclear Power Plant, LLC (the licensee) submitted a request for changes to the R.E. Ginna Nuclear Power Plant (Ginna) Technical Specifications (TSs). The requested changes would revise, on a one-time basis, TS 5.5.15, “Containment Leakage Rate Testing Program,” to extend the current interval between the containment integrated leakage rate tests (Type A tests) from 10 years to no more than 15 years.

The June 8 and August 31, 2005, letters 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 33217).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” Option B, “Performance Based Requirements,” requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. Ginna TS 5.5.15, “Containment Leakage Rate Testing Program,” requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, “Performance-Based Containment Leak-Test Program,” dated September 1995. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) Report NEI 94-01, Revision 0, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J,” dated July 26, 1995.

A Type A test is an integrated (overall) leakage rate test (ILRT) of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Ginna have been successful, so the current interval requirement is 10 years.

Under the current schedule, the next Type A test for Ginna would be performed during the 2006 Refueling Outage scheduled for May 31, 2006. The licensee has proposed a change to TS 5.5.15 that would add an exception from the guidelines of RG 1.163 and NEI 94-01, Revision 0, regarding the Type A test interval. Specifically, the exception would state that the first Type A test performed after the May 31, 1996, Type A test shall be performed by May 31, 2011, which is 15 years after the last successful Type A test.

The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

3.0 TECHNICAL EVALUATION

3.1 Background

Ginna is a pressurized-water reactor (PWR) with a large prestressed-concrete primary containment. The containment consists of the concrete containment structure, its steel liner, and the penetrations through this structure. The steel liner and its penetrations establish the leakage limiting boundary of the containment. The integrity of the penetrations is verified through Type B and Type C local leak rate tests (LLRT), as required by Appendix J to 10 CFR Part 50, and the overall integrity of the containment structure is verified through a Type A ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment structure at the design-basis accident (DBA) pressure.

3.2 Licensee's Basis for Proposed Change to Ginna TSs

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the March 10, 2005, application for license amendment. Additional analysis and information was provided by the licensee in letters, dated June 8, 2005, and August 31, 2005. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Research Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Study TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for Ginna early in the plant's life required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak, that was detectable only by a Type A test, goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and boiling-water reactor plants represented in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the Nuclear Regulatory Commission (NRC) staff has issued RG 1.174 on the use of PRA in evaluating risk-informed changes to a plant's licensing basis. The licensee proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early-release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk are based on a change in test frequency from three tests in 10 years (the test frequency under Appendix J, Option A) to one test in 15 years. This bounds the impact of extending the test frequency from one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk is estimated to be approximately 0.2 person-rem per year. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be about 5.2×10^{-7} per year based on Revision 4.3 of the Ginna PRA, which includes internal events, internal floods, fires, and shutdown events. There is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the visual examinations of the containment surfaces (as identified in American Society of

Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsections IWE/IWL). Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1 x 10^{-7} per year, and is included in the above LERF estimate.

When the calculated increase in LERF is in the range of 10^{-7} per year to 10^{-6} per year, applications are considered if the total LERF is less than 10^{-5} per year. The licensee estimates that the baseline LERF is 6.44 x 10^{-6} per year. The total LERF including the requested change is 6.96 x 10^{-6} per year, which meets the total LERF criteria. The NRC staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimated the change in the conditional containment failure probability to be an increase of approximately one percentage point for the cumulative change of going from a test frequency of three in 10 years to one in 15 years. The NRC staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

3.3 <u>Engineering Mechanics Evaluation and Inservice Inspection (ISI)</u>

Because the leakage rate testing requirements (ILRT and LLRTs) of Option B in Appendix J to 10 CFR Part 50 and the containment ISI requirements mandated by 10 CFR 50.55a, "Codes and standards," complement each other in ensuring the leak-tightness and structural integrity of the containment, the NRC staff, from its review of Type A test interval extension applications for other plants, had identified the following five general areas that the licensee was requested to address in relation to the ISI of the containment:

1. Since the submittal did not include sufficient description or summary of the containment ISI program being implemented at the plant, provide a description of the ISI methods that provide assurance that, in the absence of a containment ILRT for 15 to 20 years, the containment structural and leak-tight integrity will be maintained.

2. Subsection IWE-1240 of ASME Code, Section XI requires licensees to identify the containment surface areas requiring augmented examinations. Provide the locations of the steel containment (or concrete containment liner) surfaces that have been identified as requiring augmented examination and a summary of the findings of the examinations performed.

3. For the examination of penetration seals and gaskets, and examination and testing of bolted connections associated with the primary containment pressure boundary

(Examination Categories E-D and E-G), the licensee requested relief from the requirements of the code. As an alternative, the licensee proposed to examine the above items during the leak-rate testing of the primary containment. Option B of Appendix J for Type B and Type C testing (per NEI 94-01 and RG 1.163), and the ILRT extension requested in this amendment for Type A testing, provide flexibility in the scheduling of these inspections. Discuss your schedule for examination and testing of seals, gaskets, and bolted connections that provide assurance regarding the integrity of the containment pressure boundary.

4. In some cases, the stainless steel bellows were found to be susceptible to trans-granular stress-corrosion cracking, and the leakage through these bellows is not readily detectable by the Type B testing (see Information Notice 92-20). If applicable, provide information regarding your plans for inspection and testing of the bellows, and how their performance has been factored into the risk assessment of containment leakage to support the proposed TS change.

5. Inspections of some reinforced concrete and steel containment structures have identified degradation of an uninspectable (embedded) side of the steel liner of the primary containment. These degradations cannot be found by visual (i.e., VT-1 of VT-3) examinations unless they are through the thickness of the shell or liner, or when 100 percent of the uninspectable surfaces are periodically examined by ultrasonic testing. Discuss how potential leakage under high pressure during core damage accidents is factored into the risk assessment related to the extension of the ILRT.

The licensee provided its response to the above items in its March 10, 2005, application. The NRC staff evaluated the licensee's responses as discussed below.

1. In addressing the first issue, the licensee stated that containment leak tight integrity is also verified through periodic ISIs conducted in accordance with the requirements of the ASME Code, Section XI, 1992 Edition through the 1992 Addenda; specifically, subsections IWE and IWL, as modified and supplemented by 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix). The licensee implemented the containment program at Ginna in September 1998. This program outlines the first IWE/IWL ISI interval requirements and was formally included in the ASME Code, Section XI ISI Program. The first IWE/IWL ISI interval ends in September 2008. No evidence of significant degradation was found. The licensee also stated that continued implementation of the IWE/IWL ISI Program will provide ongoing confirmation that the effects of aging for containment structure concrete components remain inactive at Ginna Station and that their intended functions will be maintained during the period of both the extended ILRT test interval. Additionally, plant operating experience has shown that borated water spills in containment have the potential to impact the containment liner. Accordingly, the boric acid corrosion program is also credited with assessing and managing the loss of material in the containment liner. In addition, the licensee stated that inspections in accordance with the requirements of RG 1.163, Position C.3, have been conducted during the 10-year interval starting with the completion of the last ILRT and associated Structural Integrity Test (SIT) on May 31, 1996, and were completed on the following dates: November 14, 1997, April 8, 1999, and October 13, 2000. The next inspections are scheduled to be performed during the 2005 and 2008 refueling outages. In NUREG-1786 "Safety Evaluation Report to the License Renewal of R.E. Ginna Nuclear

Power Plant," the NRC staff identified that the licensee should develop a periodic functional test that would verify the containment functionality at the location of the containment support, and suggested the performance of two or three SITs during the extended period of operation. In a letter dated July 30, 2003, the licensee committed to perform two SIT inspections, one in 2015 and one in 2026.

In addition, the licensee stated that Appendix J, Type B and Type C local leak tests are not affected by the change to the Type A test frequency. Also, a structure monitoring program was developed in response to the requirements of the Maintenance Rule (see 10 CFR 50.65) and the License Renewal Rule (see 10 CFR Part 54), which provide reasonable assurance that the effects of aging will be adequately managed for the containment structure concrete components. Based on the above discussion, the NRC staff finds that the containment ISI program and other programs being implemented at Ginna will provide assurance that, in the absence of a containment ILRT for 15 years, the containment structural and leak-tight integrity will be maintained.

2. Regarding the application of an augmented examination, the licensee stated that plant-specific operating experience and recent maintenance and corrective action documents identified only one nonconforming condition at the moisture barrier (caulking), which protects the inaccessible portion of the containment steel liner from corrosion, that was discovered during the ISI performed in 2000. Evidence of minor surface corrosion was present in the area with nonconforming caulking detail. As result of this discovery, the configuration of the moisture barrier was inspected around the entire circumference of the containment and verified to be intact with no visible gaps or discontinuities. The licensee also stated that additional inspections of the liner were performed during the 2002 refueling outage. Measurements taken verified that no loss of liner thickness had occurred. Additional inspections of the moisture barrier and liner are planned during the second and third periods of the fourth ISI interval, which commenced on January 1, 2000. Ginna has committed to the performance of visual inspections and ultrasonic testing (UT) thickness measurements of the containment liner during the 2005 refueling outage. During the 2003 refueling outage, boric acid corrosion was found in the 'A' containment sump. The area of corrosion was cleaned, UT inspected, repaired, and repainted. On the basis of the above discussion, the NRC staff finds that the licensee's response is reasonable and acceptable.

3. With regard to the issue related to the ISI of seals, gaskets and pressure-retaining bolted connections, the licensee stated that for the fourth ISI interval program, there were two relief requests approved by the NRC on August 16, 1999, that defer testing to the Containment Leakage Rate Testing Program as an alternative examination. These are Relief Request Nos. RR-9, "Containment Inspection Seals and Gaskets," and RR-11, "Containment Inspection Bolt Torque or Tension Testing." The licensee's proposed one-time test interval extension request applies only to the 10 CFR Part 50, Appendix J, Type A integrated leak rate test that is currently on a 10-year interval pursuant to Appendix J, Option B, Performance Based Requirements. Appendix J, Type B and Type C tests are performed in the 10-year interval. On this basis, the NRC staff finds that the schedule for examination of the seals, gaskets, and containment pressure retaining bolting will continue to provide reasonable assurance that the integrity of the containment pressure boundary will be maintained.

4. The licensee stated that there are no penetration bellows at Ginna that perform a containment isolation function. The bellows are single-ply, American Society of Testing and Materials (ASTM) A240, Type 304 stainless steel and function to accommodate lateral and axial pipe displacements. Prior to the performance of Type A testing, the penetration bellows are aligned to their associated mechanical manifolds to permit the monitoring of the containment primary barrier welds. The pressure gauge for each manifold will be monitoring a group of penetrations. The licensee also stated that for those manifolds exhibiting pressure build-up during the ILRT, the penetrations served by those manifolds will be individually checked upon completion of the ILRT and the leakage located and the leak rate determined. On the basis of the above discussion, the item related to NRC Information Notice 92-20 is adequately addressed.

5. Regarding the inaccessible areas of the containment liner for which degradations cannot be detected by visual examinations, the licensee stated that Attachment 1 to its March 10, 2005, application provided an evaluation considering potential corrosion impacts within the framework of the ILRT interval extension risk assessment. The results of the risk assessment indicated that the ILRT interval extension has a minimal impact on plant risk. The licensee also stated that a series of parametric sensitivity studies regarding the potential age-related corrosion effects on the steel liner also indicated that, even with very conservative assumptions, its conclusions from the original analysis would not change. The analysis also provides a discussion on the effects ILRT interval extension would have on the total LERF (internal and external events), as described in Section 3.2 of this safety evaluation. From its review of the licensee’s submittal, the NRC staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and is, therefore, acceptable.

In addition to answering the above general questions, the licensee also stated that two sections of the dome on the Ginna containment building were demolished to allow the replacement of steam generators in 1996. That construction period provided a window of opportunity to perform an in-depth aging investigation on the structure after almost 30 years of services. A visual inspection of the demolished area was conducted. The licensee stated that the general condition of the inspected area was excellent. No sign of degradation or damage was detected. The study showed that the concrete dome of the containment building after almost 30 years of being exposed to the environment has not degraded and the effect of aging has been insignificant on this particular structure.

3.4 Summary

On the basis of its review of the information provided by the licensee in its application and supplemental letters, the NRC staff finds that: (1) the increase in predicted risk due to the proposed change is within the acceptance guidelines, while maintaining the defense-in-depth philosophy, of RG 1.174, (2) the structural integrity of the containment vessel is verified through the periodic ISIs that are already conducted as required by Subsection IWE of the ASME Code, Section XI, and (3) the integrity of the penetrations and containment isolation valves will periodically be verified through Type B test as required by 10 CFR Part 50, Appendix J. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities, if any, in accordance with Article IWE-5000 of the ASME Code, Section XI. Therefore, the NRC staff finds the proposed one-time extension of the ILRT test interval (ILRT to be conducted no later than May 31, 2011) and the associated changes to TS 5.5.15 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 33217). 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. Pulsipher Z. Cruz-Perez
R. Palla T. Cheng

Date: December 8, 2005