Commission regulations, and individual radiation exposure reports which are available to him.

A copy of the final supporting statement may be viewed free of charge at the NRC Public Document Room, 2120 L Street, NW (lower level), Washington, DC. OMB clearance requests are available at the NRC worldwide web site (http://www.nrc.gov) under the FedWorld collection link on the home page tool bar. The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Comments and questions should be directed to the OMB reviewer by June 11, 1998: Erik Godwin, Office of Information and Regulatory Affairs (3150–0044), NEOB–10202, Office of Management and Budget, Washington, DC 20503.

Comments can also be submitted by telephone at (202) 395–3084.

The NRC Clearance Officer is Brenda Jo. Shelton, 301–415–7233.

Dated at Rockville, Maryland, this 6th day of May 1998.

For the Nuclear Regulatory Commission.

Brenda Jo. Shelton,

NRC Clearance Officer, Office of the Chief Information Officer.

[FR Doc. 98–12527 Filed 5–11–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–317, 50–318, and 72–8]

In the Matter of Baltimore Gas Electric Company (Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and the Independent Spent Fuel Storage Installation; Order Terminating the Effectiveness of the Approval of the Transfer of Licenses for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 and the Independent Spent Fuel Storage Installation

I

Baltimore Gas and Electric Company (BGE) is the licensee for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, and the associated Independent Spent Fuel Storage Installation. BGE has the exclusive responsibility for the construction, operation, and maintenance of Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 and the Independent Spent Fuel Storage Installation (ISFSI), as reflected in Operating License Nos. DPR–53, DPR–69 and Material License No. SNM–2505, issued on July 31, 1974, and November 30, 1976, and November 25, 1992, respectively, by the U.S. Nuclear Regulatory Commission (NRC). The facilities are located on the western shore of the Chesapeake Bay, in Calvert County, Maryland.

II

By Order dated October 18, 1996, the Nuclear Regulatory Commission (the Commission or NRC) approved the proposed transfer of Operating Licenses Nos. DPR–53 and DPR–69 for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2, and Material License No. SNM–2505 for the Calvert Cliffs ISFSI from BGE to Constellation Energy Corporation. The approval was given in response to an application filed by BGE dated April 5, 1996, for consent under Section 50.80 and 72.50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.80 and 10 CFR 72.50). By its terms, the Order of October 18, 1996, would become null and void if the transfer of the licenses was not consummated by December 31, 1997, unless on application and for good cause shown, such date was extended by the Commission.

By letter dated November 21, 1997, BGE submitted a request for an extension of the effectiveness of the Order of October 18, 1996, such that approval of the transfer would remain effective until December 31, 1998. According to this submittal, all of the necessary regulatory approvals had been obtained to permit the consummation of the merger between BGE and Potomac Electric Power Company, resulting in Constellation Energy Corporation. BGE asserted, however, that the Maryland and District of Columbia Public Service Commissions attached conditions to their approvals that were inconsistent with the respective merger approval applications. The companies proposing to merge filed joint requests with the Maryland and District of Columbia Commissions for rehearing of their original orders approving the merger. According to BGE, an intervenor in the Maryland case appealed the Maryland Commission's Order approving the merger to the Circuit Court in Baltimore County, and this appeal delayed the expected merger process. On December 17, 1997, the Commission issued an Order providing that the effectiveness of the Order of October 18, 1996, approving the transfer of the licenses described herein was extended such that if the subject transfer of licenses was not consummated by December 31, 1998, the Order of October 18, 1996, would become null and void.

By letter dated January 30, 1998, however, BGE informed the NRC that on December 18, 1997, BGE and the Potomac Electric Power Company (PEPCO) mutually agreed to terminate the proposed merger. In addition, BGE and PEPCO requested, in light of the termination of the merger, that approval of the transfer of licenses be canceled.

III

Upon consideration of BGE's letter dated January 30, 1998, and the termination of the proposed merger, the Commission has determined that the approval of the transfer of the licenses for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, and the ISFSI, should be withdrawn. Accordingly, pursuant to Sections 161b and 161i of the Atomic Energy Act, as amended, 42 U.S.C. §§ 2201(b) and 2201(i), It is hereby ordered that the approval of the transfer of the licenses described herein is immediately withdrawn, and the Orders dated October 18, 1996, and December 19, 1997 are null and void.

This Order is effective upon issuance.

For further details, with respect to this action, see the letter dated January 30, 1998, from BGE which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, and at the local public document room located at the Calvert County Library, Prince Frederick, Maryland 20678.

Dated at Rockville, Maryland, this 30th day of April 1998.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

Carl J. Paperiello,

Director, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 98–12524 Filed 5–11–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–244]

Rochester Gas and Electric Corporation; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. DRP–18 issued to Rochester Gas and Electric Corporation (the licensee) for operation of the R.E. Ginna Nuclear Power Plant located in Wayne County, New York.

The proposed amendment would revise the Ginna Station Improved Technical Specifications (ITS) to reflect a planned modification to the spent fuel pool (SFP) storage racks. Specifications associated with SFP boron concentration, fuel assembly storage, and maximum limit on the number of fuel assemblies which can be stored in the SFP would be revised.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The design basis events considered for the spent fuel pool include both external events and postulated accidents in the pool. The external events considered are tornado missiles and seismic events. The evaluation of the postulated impact of a tornado missile is detailed in Sections 3, 4, and 6 of Reference 1 [see application dated March 31, 1997]. The structural evaluation indicates that there are no gross distortions of the racks or any adverse effects upon plant structures or equipment. The radiological consequences of this event indicate that offsite doses are ''well within'' the 10 CFR 100 limits.

The structural evaluation is detailed in Section 3 of Reference 1 [see application dated March 31, 1997]. Current state of the art methods are used in the structural analysis. The evaluation of the storage racks is based on a conservative interpretation of the ASME [American Society of Mechanical Engineers] Boiler and Pressure Vessel Code. The evaluation of the spent fuel pool is based on a conservative interpretation of requirements set forth in the American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures, and American Institute of Steel Construction, Specification for Structural Steel Buildings. The spent fuel storage system was designed to meet all applicable structural criteria for normal (Level A), upset (Level B), and faulted (Level D) conditions as defined in NUREG–0900, SRP [Standard Review Plan] 3.8.4, Appendix D. The following loadings were considered: dead weight, seismic, thermal, stuck fuel assembly, drop of a fuel assembly, and tornado missile impact. Load combinations were performed in accordance with SRP 3.8.4, Appendix D. Given the evaluated seismic events, the changes in the final position of the racks are small as compared to the initial position prior to the seismic event. The maximum closure of gaps is such that no significant changes in gaps results during any single seismic event. Furthermore, the combined gap closures resulting from a combination of 5 OBEs [Operating Basis Earthquakes] and 1 SSE [Safe Shutdown Earthquake] show that there are no rack-to-rack or rack-to-wall impacts. These evaluations conclude that under these postulated events, the stored fuel assemblies are maintained in a stable, coolable geometry, and a subcritical configuration.

As described in the bases for LCO [Limiting Condition for Operation] 3.7.12 and 3.7.13, the postulated accidents in the spent fuel pool are divided into two categories. The first are those involving a loss of cooling in the spent fuel pool. The thermal-hydraulic analysis for the maximum expected decay heat loads is described in Section 5 of Reference 1 [see application dated March 31, 1997]. The proposed modification does not change the configuration of the available spent fuel cooling systems, the limiting design conditions for maximum decay heat load which occurs during a full core offload, or the existing requirement to maintain pool temperature below 150 °F. Utilizing the three available spent fuel cooling systems, Ginna Station maintains full redundancy during high heat load conditions. The decay heat load to the spent fuel pool is maintained within the capacity of the operating cooling system by appropriately delaying fuel offload from the reactor. Should a failure occur on the operating cooling system, the resulting heat rates allow sufficient time to place a standby cooling system in service before the pool design limit temperature is exceeded. Increases in spent fuel pool temperature, with the corresponding decrease in water density and void formation from boiling, will result in a decrease in reactivity due to the decrease in moderation effects. In addition, the analysis demonstrates that the storage rack geometry and required fuel storage configurations result in a k_{eff} [less than or equal to] .95 assuming no soluble boron allowing for the potential of makeup to the pool with unborated water if credit is taken in Region 2 for minimal availability of boraflex panels installed on the storage rack. (Note that concerns with boraflex degradation are discussed later in this evaluation).

The second category is related to the movement of fuel assemblies and other loads above the spent fuel pool. The limiting accident with respect to reactivity is the fuel handling accident which is analyzed in Section 4 of Reference 1 [see application dated March 31, 1997]. For both the incorrectly transferred fuel assembly (placed in an unauthorized location) or a dropped fuel assembly, the positive reactivity effects resulting are offset by the negative reactivity from the required minimum soluble boron concentration. The resulting k_{eff} is shown to be less than 0.95 if credit is taken in Region 2 for minimal availability of boraflex panels installed on the storage racks. The radiological consequences of a fuel assembly drop remain as described in Section 15.7.3 of the UFSAR [updated final safety analysis report] and as discussed in Section 6 of Reference 1 [see application dated March 31, 1997]. Loads in excess of a fuel assembly and its handling tool are administratively prohibited from being carried over spent fuel. There are no changes anticipated for either the fuel handling equipment of the auxiliary building overhead crane due to the proposed modification to the fuel storage racks. The modification is scheduled for the Year 1998 to be performed while Ginna Station is operating. Movement of heavy loads around the spent fuel pool are controlled by the requirements of NUREG–0612 and the regulatory guidelines set forth in NRC Bulletin 96–02 (see Section 3 of Reference 1 [see application dated March 31, 1997]). Spent fuel casks and storage racks (during removal and installation) will be moved using the auxiliary building crane and lifting attachments satisfying the single failure proof criteria of NUREG–0554, obviating the need to determine the consequences for this accident.

Due to boraflex degradation within the spent fuel pool, credit must be temporarily taken for soluble boron to maintain k_{eff} [less than or equal to] 0.95. There is no increase in the probability of a loss of spent fuel pool cooling or fuel handling accident as a result of crediting soluble boron. The spent fuel pool is normally maintained at a boron concentration level greater than that proposed, including during fuel movement. Therefore, there is no effect on plant systems or spent fuel pool activities than which are currently in effect. The proposed boron concentration level is also equivalent to that required by LCO 3.9.1 during MODE 6 such that no boron dilution event is expected to occur within the pool during refueling operations when the reactor coolant system and spent fuel pool are hydraulically coupled.

Crediting soluble boron does not increase the consequences of an accident. As described in the bases for LCO 3.7.12, increases in spent fuel pool temperature, with the corresponding decrease in water density and void formation from boiling, will generally result in a decrease in reactivity due to the decrease in moderation effects. The only exception are temperature bands where positive reactivity is added as a result of the high boron concentration. This effect is bounded by the reactivity added as a result of a misloaded fuel assembly. With respect to the more limiting dropped fuel assembly accidents, boraflex neutron absorber panels were originally assumed in the criticality analysis. Requiring a high concentration of soluble boron in place of boraflex panels ensures that the spent fuel pool remains subcritical with k_{eff} [less than or equal to] 0.95 for these accidents. Fuel assembly movement will continue to be controlled in accordance with plant procedures and LCO

3.7.13 which specifies limits on fuel assembly storage locations. Periodic surveillances of boron concentration will be required every 7 days with level verified every 7 days during fuel movement per LCO 3.7.11. Due to the large inventory within the spent fuel pool, dilution of the soluble boron within the pool is very unlikely without being detected by operations personnel during auxiliary operator rounds or available level detection systems. There is also a large margin between the required boron concentration to maintain the pool subcritical k_{eff} [less than or equal to] 0.95 and the proposed value (approximately 900 ppm).

Based on the above, it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously analyzed.

2. Operation in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed modification does not alter the function of any system associated with spent fuel handling, cooling or storage. The proposed changes do not involve a different type of equipment or changes in methods governing normal plant operation. The additional restrictions placed on the acceptable storage locations for spent fuel are consistent with the type of restriction that previously existed. The potential violation of these restrictions (incorrectly transferred fuel assembly) are analyzed as discussed above. The rerack design, analysis, fabrication, and installation meet all the appropriate NRC regulatory requirements, and appropriate industry codes and standards.

Crediting soluble boron within the spent fuel pool in place of boraflex neutron absorber panels does not create the possibility of a new or different kind of accident since the spent fuel pool is normally maintained with high boron concentrations. Assuming a boron dilution event to the level required to reach k_{eff} [less than or equal to] 0.95 conditions within the spent fuel pool would require either overfill of the pool or a controlled feed and bleed process with unborated water. In both cases, greater than 105,000 gallons of unborated water would be required to reach $k_{eff} > 0.95$. There is no source of unborated water of this size available to reach the spent fuel pool under procedural control or via a pipe break other than a fire water system pipe break or SW leak through the spent fuel pool heat exchangers. However, there are numerous alarms available within the control room to indicate this condition including high spent fuel pool water level and sump pump actuations within the residual heat removal pump pit (lowest location in the Auxiliary Building). Auxiliary operators also perform regularly scheduled tours within the Auxiliary Building. This provides sufficient time to terminate the event such that there is no credible spent fuel pool dilution accident.

Based on the above, the change does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in the margin of safety.

The Licensing Report enclosed as Reference 1 [see application dated March 31, 1997] addresses the following considerations: nuclear criticality, thermal-hydraulic, and mechanical, material, and structural. Results of these evaluations demonstrate that the changes associated with the spent fuel reracking does not involve a significant reduction in the margin of safety as summarized below:

Nuclear Criticality

The established regulatory acceptance criterion is that k_{eff} be less than or equal to 0.95, including all uncertainties at the 95/95 probability/confidence level, under normal and abnormal conditions. The methodology used in the evaluation meets NRC requirements, and applicable industry codes, standards, and specifications with credit taken in Region 2 for the previously installed boraflex panels. In addition, the methodology has been reviewed and approved by the NRC in recent nuclear criticality evaluations. Specific conditions which were evaluated include misloading of a fuel assembly, drop of a fuel assembly (shallow, deep drops, and side drops), pool water temperature effects, and movement of racks due to seismic events. Results described in Section 4 of Reference 1 [see application dated March 31, 1997] document that the criticality acceptance criterion is met for all normal and abnormal conditions.

Thermal-Hydraulic

Conservative methods and assumptions have been used to calculate the maximum temperature of the fuel and the increase of the bulk pool water temperature in the spent fuel pool under normal and abnormal conditions. The methodology for performing the thermal-hydraulic evaluation meets NRC regulatory requirements. Results from the thermal-hydraulic evaluation show that the maximum temperature at the hottest fuel assembly, intact or consolidated canister, is less than the temperature for nucleate boiling condition. The effects of cell blockage on the maximum temperature of intact fuel and consolidated canisters were evaluated. Results described in Section 5 of Reference 1 [see application dated March 31, 1997] show that adequate cooling of the intact or consolidated fuel is assured. In all cases, the existing spent fuel pool cooling system will maintain the bulk pool temperature at or below 150 °F by delaying core offload from the reactor.

Mechanical, Material, and Structural

The primary safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all normal and abnormal loads. Abnormal loadings which have been considered in the evaluation are: seismic events, the drop of a fuel assembly, the impact of a tornado missile, a stuck assembly, and the drop of a heavy load. The mechanical, material, and structural design of the new spent fuel racks is in accordance with NRC regulatory requirements (including the NRC OT Position dated April 14, 1978, [NRC letter to all power reactor licensees dated April 14, 1978] and addendum dated January 18, 1979), and applicable industry standards. The rack materials are compatible with the spent fuel pool environment and fuel assemblies. The material used as a neutron absorber (borated stainless steel) has been approved by the American Society for Testing and Materials (ASTM), and licensed previously by the NRC for use as a neutron absorber at Indian Point 3, Indian Point 2, and Millstone 2. The structural evaluation presented in Section 3 of Reference 1 [see application dated March 31, 1997] documents that the tipping or sliding of the free-standing racks will not result in rack-to-rack or rack-to-wall impacts during seismic events. The spent fuel assemblies will remain intact and the criticality criterion of k_{eff} [less than or equal] to 0.95 is met if credit is taken in Region 2 for previously installed boraflex panels.

Soluble boron within the spent fuel pool provides a significant negative reactivity such that k_{eff} is maintained [less than or equal to] 0.95. The proposed surveillance frequency will ensure that the necessary boron concentration is maintained. A boron dilution event which would remove the soluble boron from the pool has been shown to not be credible.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of

Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By June 11, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Rochester Public Library, 115 South Avenue, Rochester, New York 14610. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to Nicholas S. Reynolds, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated March 31, 1997, supplemented June 18, 1997, October 10, 1997, October 20, 1997, November 11, 1997, December 22, 1997, January 15, 1998, January 27, 1998, March 30, 1998, April 23, 1998, and April 27, 1998, which are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Rochester Public Library, 115 South Avenue, Rochester, New York 14610. This notice supersedes the March 31, 1997, application published on April 30, 1997 (62 FR 23502) in its entirety.

Dated at Rockville, Maryland, this day of May 1998.

For the Nuclear Regulatory Commission.

Guy S. Vissing,

Senior Project Manager, Project Directorate I–1, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 98–12526 Filed 5–11–98; 8:45 am]

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