and modifications without shutting down both Peach Bottom units.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed action and based on the information presented in the licensee's application, concludes that the proposed extension of the EDG's AOT in conjunction with the availability of the Conowingo line, will not increase the probability of initiating events leading to a design basis accident. The additional reliability of the offsite source afforded by the Conowingo line would improve the potential for mitigating loss-of-offsite power events. Consequently, the consequences of accidents would not be significantly increased, nor would the post-accident radiological releases be greater than previously determined.

The proposed action would not otherwise affect radiological plant effluents. Accordingly, the Commission concludes that there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action (extending EDG AOTs) does not affect nonradiological plant effluents and has no other environmental impact. Accordingly, the Commission concludes that there are no significant nonradiological environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

Since the Commission has concluded there is no measurable environmental impact associated with the proposed action, any alternatives with equal or greater environmental impact need not be evaluated. As an alternative to the proposed action, the staff considered denial of the proposed action. Denial of the application would result in no change in current environmental impacts. The environmental impacts of the proposed action and the alternative action are similar.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for the Peach Bottom Atomic Power Station, Units 2 and 3, dated April 1973.

Agencies and Persons Consulted

In accordance with its stated policy, on July 24, 1995, the staff consulted with the Pennsylvania State official, Stan Maingi, of the Pennsylvania Department of Environmental Resources, regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated April 7, 1994, as supplemented by letters dated June 2, and September 6, 1994, and June 16, and July 13, 1995, 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 Government Publications Section, State Library of Pennsylvania, (Regional Depository) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania.

Dated at Rockville, MD, this 4th day of August 1995.

For the Nuclear Regulatory Commission.

John F. Stolz,

Director, Project Directorate I–2, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–19764 Filed 8–9–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–219]

GPU Nuclear Corporation, Oyster Creek Nuclear Generating Station; Issuance of Partial Director's Decision Under 10 CFR § 2.206

Notice is hereby given that the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC) has denied in part a Petition, dated September 19, 1994, and supplemented December 13, 1994, submitted by Oyster Creek Nuclear Watch, Reactor Watchdog Project, and Nuclear Information and Resource Service (Petitioners). The Petition requested that the NRC take action regarding the Oyster Creek Nuclear Generating Station (OCNGS) pursuant to 10 C.F.R. § 2.206.

The September 19, 1994, Petition requests that the NRC (1) immediately suspend the OCNGS operating license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the Licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the Licensee has analyzed and mitigated any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling-water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements, and to promptly take appropriate mitigative action if the units are not in compliance.

The December 13, 1994, supplemental Petition requests that the NRC: (1) suspend the license of the OCNGS until the Petitioners' concerns regarding cracking are addressed, including inspection of all reactor vessel internal components and other safety-related systems susceptible to intergranular stress corrosion cracking (IGSCC) and completion of any and all necessary repairs and modifications; (2) explain discrepancies between the response of the NRC staff dated October 27, 1994, to the Petition of September 19, 1994, and the time-to-boil calculations for the FitzPatrick plant; (3) require the GPU Nuclear Corporation to produce documents for evaluation of the time-to-boil calculation for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class 1E systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the Petition; and (6) treat the Petitioners' letter of December 13, 1994, as a formal appeal of the denial of the Petitioners' request of September 19, 1994, to immediately suspend the OCNGS operating license.

The Director of the Office of Nuclear Reactor Regulation has denied Requests (1) and (2) of the September 19, 1994, Petition and Request (1) of the December 13, 1994, supplemental Petition to suspend the operating license of the OCNGS until the Licensee inspects and repairs, modified, or replaces all safety-class reactor internal component parts subject to embrittlement and intergranular stress corrosion cracking. The reasons for this denial are explained in the ''Partial Director's Decision Under 10 CFR § 2.206'' (DD–95–18), the complete text of which follows this notice, and 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 for the Oyster Creek Nuclear Generating Station located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753. A decision regarding Requests (3), and (4) of the September 19, 1994 Petition, and Requests (2), (3), and (4), of the December 13, 1994, supplemental Petition will be issued under separate cover upon completion of the NRC staff's review.

A copy of this Partial Director's Decision will be filed with the Secretary of the Commission for review in accordance with 10 CFR 2.206(c). As provided in that regulation, the Decision will constitute the final action of the Commission 25 days after the date of the issuance of the Decision, unless the Commission, on its own motion, institutes a review of the Decision within that time.

Dated at Rockville, Maryland this 4th day of August 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

Appendix A—Partial Director's Decision Under 10 CFR § 2.206 (DD95–18)

I. Introduction

By letter dated September 19, 1994, Reactor Watchdog Project, Nuclear Information and Resource Service (NIRS), and Oyster Creek Nuclear Watch (Petitioners), submitted a Petition pursuant to Section 2.206 of Title 10 of the *Code of Federal Regulations* (10 C.F.R. § 2.206), requesting that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to the Oyster Creek Nuclear Generating Station (OCNGS), operated by the GPU Nuclear Corporation (GPUN or the Licensee). By letter dated December 13, 1994, Petitioners supplemented the Petition.

The September 19, 1994, Petition requests that the NRC (1) immediately suspend the OCNGS operating license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking, (2) immediately suspend the OCNGS operating license until the Licensee submits an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components, (3) immediately suspend the OCNGS operating license until the Licensee has analyzed and mitigated any areas of noncompliance with regard to irradiated fuel pool cooling as a single-unit boiling-water reactor (BWR), and (4) issue a generic letter requiring other licensees of single-unit BWRs to submit information regarding fuel pool boiling in order to verify compliance with regulatory requirements, and to promptly take appropriate mitigative action if the unit is not in compliance.

The December 13, 1994, supplemental Petition requests that the NRC: (1) suspend the license of the OCNGS until the Petitioners' concerns regarding cracking are addressed, including inspection of all reactor vessel internal components and other safety-related systems susceptible to intergranular stress corrosion cracking (IGSCC) and completion of any and all necessary repairs and modifications; (2) explain discrepancies between the response of the NRC staff dated October 27, 1994, to the Petition of September 19, 1994, and the time-to-boil calculations for the FitzPatrick plant; (3) require GPUN to produce documents for evaluation of the time-to-boil calculation for the OCNGS irradiated fuel pool; (4) identify redundant components that may be powered from onsite power supplies to be used for spent fuel pool cooling as qualified Class 1E systems; (5) hold a public meeting in Toms River, New Jersey, to permit presentation of additional information related to the Petition; and (6) treat the Petitioners' letter of December 13, 1994, as a formal appeal of the denial of the Petitioners' request of September 19, 1994, to immediately suspend the OCNGS operating license.

The September 19, 1994, Petition sought relief concerning safety-class reactor internal components based on the following premises: (a) the core shroud in General Electric BWRs is vulnerable to age-related deterioration; (b) 12 domestic and foreign BWR owners have found extensive cracking on welds of the core shroud; (c) only 10 of 36 U.S. BWR owners have inspected their core shrouds and 9 of the 10 core shrouds had cracks; (d) 19 of 25 selected BWR internal components are susceptible to stress corrosion cracking and 6 of 19 are susceptible to irradiation-assisted stress corrosion cracking; (e) as the oldest operating General Electric Mark I BWR and the third oldest operating reactor in the United States, OCNGS has been subjected to the longest period of operational conditions that cause embrittlement and cracking; (f) the BWR Owners Group (BWROG) stated that cracking of the core shroud is a warning signal that additional safety-class reactor internals are increasingly susceptible to age-related deterioration; (g) cracking of any single part or multiple components jeopardizes safe operation of that nuclear station; (h) Oyster Creek did not inspect for core shroud cracking prior to the current refueling outage and other safety-class reactor internals have not been adequately inspected for cracking; and (i) a safety analysis has not been performed on the potential synergistic effects of multiple-component cracking.

The September 19, 1994, Petition also sought relief concerning fuel pool cooling design deficiencies, based on the following premises: (a) various design defects in BWR fuel pool cooling systems pose a significant increase in risk to the public safety and violate 10 CFR 50.59; 10 CFR Part 50, Appendix A, Criterion 63; 10 CFR Part 50, Appendix B, Criterion III; and Regulatory Guides 1.13, 1.89, and 1.97; (b) OCNGS is a single-unit facility with no adjacent units to rely upon in the event that a design-basis event were to disable the fuel pool cooling system; and (c) OCNGS has not docketed any material with regard to BWR design deficiencies identified in the 10 CFR Part 21 Report of Substantial Safety Hazard (November 27, 1992) of Messrs. Lochbaum and Prevatte, and thus OCNGS may be in violation of NRC regulatory requirements.

The Petitioners assert the following bases to support their requests in the December 13, 1994, supplemental Petition: (a) the October 27, 1994, letter of the NRC staff, acknowledging receipt of the Petition and denying the requests for immediate suspension of the operating license, failed to address concerns central to the Petition, such as the Licensee's failure to recognize that IGSCC indicates that cracking could be occurring in additional safety-class reactor internal components and the Licensee's failure to perform inspections of all safety-class components to determine whether cracking is occurring; (b) recently discovered cracking in the top guide and core plates in foreign BWRs and cracking discovered on December 8, 1994, at the New York Power Authority's (NYPA's) FitzPatrick reactor underscore the Petitioners' concern that additional safety-class components at OCNGS are degrading; (c) the Licensee did not conduct an enhanced inspection of the core plate and top guide of the OCNGS facility during the current outage, despite notification by the General Electric Rapid Information Communication Service Information Letter (GE RICSIL) 071 dated November 22, 1994; (d) the Licensee, the NRC, and the BWR Owners Group (BWORG) have failed to provide an analysis of the synergistic effects of multiple-component cracking of additional safety-class reactor internal components; (e) the time-to-boil calculation is dictated by the amount of decay heat generated and the volume of water in the fuel pool rather than the number of reactors at a site that store irradiated fuel in a separate pool; (f) NRC documents state that the time-to-boil calculation for FitzPatrick following a loss-of-coolant accident is 8 hours, and NYPA documents state that the time-to-boil calculations in two cases are 11.86 and 5.36 hours. Finally, nothing indicates that the time-to-boil calculation at OCNGS is longer than the time-to-boil calculation at the Susquehanna facility; and (g) the NRC and the licensee have failed to establish whether redundant components and power supplies to the OCNGS fuel pool cooling system have been qualified as Class 1E systems.

The Petitioners' requests that the Commission immediately suspend the OCNGS operating license were denied in my letter of October 27, 1994, to the Petitioners, because (1) OCNGS was in a refueling outage, had inspected core shroud welds, and was making structural modifications before restart of the unit to address some weld cracks found during the inspection, and (2) inspections and corrective actions recommended by General Electric Company and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for various reactor internals had been and continued to be performed by the Licensee.

The Petitioners' request for treatment of their letter of December 13, 1994, as a formal appeal of the NRC staff's denial of their request of September 19, 1994, for immediate suspension of the OCNGS operating license, was denied in my letter of April 10, 1995, to the Petitioners. The Petitioners provided no basis for revisiting the denial of their request of September 19, 1994, for immediate suspension of the license. As discussed below, the Licensee completed all ASME Code Section XI reactor vessel internal inspections and BWROG recommended inspections and took appropriate remedial action before re-start of OCNGS in December 1994. The NRC staff was also aware of the potential problem for United States BWRs raised by cracking in top guide and core plates of foreign BWRs before the restart of OCNGS. The NRC staff determined, as explained below, that cracks in these components would not adversely affect safety of the plant because of differences in the OCNGS design as compared to the affected foreign reactors.

Regarding the OCNGS spent fuel pool cooling system capability, the staff determined that the time to the onset of spent fuel pool boiling following a loss of spent fuel pool cooling during periods where the reactor vessel contains irradiated fuel at single unit BWR sites, such as OCNGS, is long enough to allow compensatory measures. The probability of a sustained loss of spent fuel pool cooling creating adverse environmental conditions that may cause failure of essential equipment is extremely low. Therefore, the staff has concluded that immediate action to address the concerns the Petitioners have identified at OCNGS is not justified. As stated in my letter of October 27, 1994, spent fuel pool safety is being reviewed generically by the staff and this review has not yet been completed.

The Petitioners' request for a public meeting was denied in my letter of April 10, 1995.[1] The issue of internals cracking has been discussed at several public meetings, including a public meeting on November 4, 1994, that a representative of NIRS attended regarding the OCNGS core shroud. With respect to spent fuel pool cooling, the staff has held several public meetings and public briefings with the Advisory Committee on Reactor Safeguards. Summaries of these public meetings are available in the NRC Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document rooms for the affected BWR plants. Transcripts of ACRS meetings are also available.

The NRC staff's review of the issues related to cracking of reactor internal components, raised by Requests (1) and (2) of the September 19, 1994, Petition, and Request (1) of the December 13, 1994, supplemental Petition, is now complete. For the reasons set forth below, the Petition is denied with respect to these requests. A Director's Decision concerning the issues related to irradiated fuel pool cooling and fuel pool boiling, raised by Requests (3) and (4) of the September 19, 1994, Petition and Requests (2), (3), and (4) of the December 13, 1994, supplemental Petition will be issued upon completion of the NRC staff's review regarding those matters.

II. Background

Intergranular stress corrosion cracking (IGSCC) of BWR internal components has been identified as a technical issue of concern by both the NRC staff and the nuclear industry. The core shroud is among the internal reactor components susceptible to IGSCC. Identification of cracking at the circumferential beltline region welds in several plants during 1993 led to the publication of NRC Information Notice (IN) 93–79, ''Core Shroud Cracking at Beltline Region Welds in Boiling-Water Reactors,'' issued on September 30, 1993. Several licensees inspected their core shrouds during planned outages in the spring of 1994 and found cracking at the circumferential welds. The NRC has closely monitored these inspection activities. Additionally, licensees have inspected other BWR reactor vessel internal components as discussed below. NRC issued IN 94–42, ''Cracking in the Lower Region of the Core Shroud in Boiling-Water Reactors,'' on June 7, 1994, and Supplement 1 to IN 94–42, on July 19, 1994, concerning cracking in the core shroud found at Dresden Unit 3 and Quad Cities Unit 1. IN 95–17, ''Reactor Vessel Top Guide and Core Plate Cracking,'' issued on March 10, 1995, concerned reactor vessel top guide and core plate cracking. The NRC has monitored Licensee inspection activities of these components at the OCNGS as discussed below.

III. Discussion

A. Petitioners request that the NRC suspend the OCNGS license until the Licensee inspects and repairs or replaces all safety-class reactor internal component parts subject to embrittlement and cracking. Nuclear power reactor licensees, including GPUN, are required by 10 C.F.R. § 50.55a to implement inservice inspection programs in accordance with the guidelines of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The scope of the inservice inspection programs for reactor pressure vessels and their internal components is prescribed by ASME Code, Section XI, Division 1, Subsections IWA and IWB. The Licensee is also required by ASME Code, Section XI, Article IWA–6000, to submit the results of these inspections to the NRC within 90 days of completion. The NRC staff performs periodic audits of licensee-implemented inservice inspection programs to determine compliance with applicable codes and regulations. These audits are documented in NRC inspection reports, which are publicly available at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the OCNGS located at the Ocean County Library, Reference

[1] In addition, the NRC staff determined, in accordance with the guidance in NRC Management Directive 8.11, ''Review Process for 10 CFR 2.206 Petitions,'' that an informal public hearing was not warranted because the Petition did not present new information or a new approach for evaluating the concerns Petitioners raised.

Department, 101 Washington Street, Toms River, NJ 08753.

The Licensee performed inspections of the OCNGS reactor vessel and its internal safety-related components in accordance with the requirements of ASME Code, Section XI, and the NRC staff has reviewed the Licensee's inservice inspection programs, as discussed below.

Cracking of the core spray piping was first detected during Licensee inspections at OCNGS in 1978, and its extent has been evaluated by the Licensee during each subsequent outage. The core spray piping was repaired in 1978 and 1980. Since that time, additional visual inspections by the Licensee have not identified any significant degradation of the piping or of the repairs made to the piping. The NRC's review of the Licensee's inspection results and disposition during the 14R outage, documented in NRC Inspection Report 50–219/92–22, dated March 19, 1993, and a letter to GPUN dated November 18, 1994, regarding the 15R inspection concluded that the Licensee inspections and dispositions of core spray system findings were appropriate.

The Licensee first detected cracking of the top guide in 1991 and has closely monitored it in successive outages. The NRC staff conducted an inspection in June 1991, and concluded that the Licensee's disposition of the top guide crack as ''acceptable as is'' was adequate. The results of the inspection were reported in NRC Inspection Report 50–219/91–21, dated August 9, 1991. During an NRC inspection conducted in December 1992 and January 1993, the NRC staff evaluated the results of a remote visual inspection of the top guide conducted by General Electric Corporation for GPUN. The staff evaluated the quality of the Licensee's visual inspection of the top guide and agreed with the Licensee's determination that the top guide was acceptable to ''use as is''. The results of the inspection were reported in NRC Inspection Report 50–219/92–22, dated March 19, 1993.

The Licensee notified the NRC staff during an October 11, 1994, telephone call that additional cracking in the top guide had been found. The Licensee also reported that cracks found in earlier inspections of the top guide had not shown any measurable growth. In addition, during the refueling outage for Cycle 15 of operation (15R refueling outage), which began in September 1994, the Licensee assessed all the cracks that had been identified to ensure they would not jeopardize the structural integrity or function of the top guide.

It should be noted that the location of the cracks that have been detected in the OCNGS top guide is different from that in the foreign reactor cited in the NIRS letter of December 13, 1994, and the subject of GE RICSIL–071. Moreover, both the top guide and the core plate at OCNGS are components of a GE BWR while the foreign plant is a non-GE BWR. Furthermore, the OCNGS core plate is bolted in place, and the top guide is restrained vertically by hold-down devices and horizontally by lateral supports. These configurations result in a highly redundant structure, and even if cracking similar to that observed in the foreign plant were to occur, it would not adversely affect the safety of the plant, and these components could still perform their safety-related functions.

The BWROG has addressed the issue of cracking in the internal components of reactor pressure vessels by recommending that BWR licensees perform inspections of various components pursuant to vendor recommendations of the General Electric Company. Among inspections recommended by the BWROG are examination of core spray spargers, core shrouds, top guides, return line nozzles, and in-core instrumentation, which in the case of OCNGS are the intermediate power range monitors. The BWROG has also formed the Boiling Water Reactor Vessels & Internals Project (BWRVIP), chaired by five nuclear industry vice presidents, to develop a proactive program to address and mitigate cracking in reactor pressure vessel internal components. NRC staff correspondence with the BWRVIP, staff evaluation of the BWRVIP generic submittals, summaries of meetings with the BWRVIP, and staff assessments of plant-specific submittals in regard to these subjects are also available to the public for review at the local public document room of each BWR plant.

The Licensee inspected the following safety-related components during the 15R refueling outage, which began in September 1994: core spray sparger and annular piping, steam dryer and separator assembly, core shroud head bolts, core support plate holddown bolts, guide rod and steam dryer support brackets, feedwater spargers, top guide assembly, four intermediate-power range monitors, one low-power range monitor, core shroud brackets, conical support to shell weld, and the core shroud. Cracking was observed on the core shroud and a steam dryer bracket, and required repairs to these components were made. Minor cracking was observed on the core spray piping, a tack weld on the keeper bolt of the feedwater spargers, and the top guide cross beams. None of these cracks would have prevented the components from performing their normal operating and postulated accident functions. These indications were dispositioned as is. The Licensee submitted results of its core shroud inspection and its core spray sparger inspection to the NRC in separate letters, both dated November 3, 1994. As a result of a conference call on January 19, 1995, the Licensee submitted a summary of the results of its inspections of reactor vessel internal components performed during the 15R refueling outage. By a letter dated March 16, 1995, in accordance with 10 CFR § 50.55a(g) and ASME Section XI, IWA 6220, (1986 Edition with no addenda), GPUN forwarded the reports of its inservice inspection activities conducted during the 15R refueling outage. In the report GPUN lists the inspections performed and discusses unacceptable indications of certain components and their disposition. Inservice inspection of reactor vessel internal components is required by the ASME Code and the licensee's inservice inspection program for future outages provides assurance that degradation of components will be detected and appropriate action will be taken. The documents discussed above are 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 Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753.

The Licensee's inspection of the OCNGS core shroud found that one of the ten circumferential welds (the H4 weld) had indications of substantial cracking. To ensure shroud integrity under all postulated accidents, the Licensee elected to install a modification, consisting of ten stabilizing tie-rods, designed to ensure that the core shroud would perform its design functions under normal operation and postulated accidents even if it were to develop 360° through-wall cracks. The NRC staff reviewed this modification and issued a safety evaluation on November 25, 1994, which concluded that the core shroud modification proposed by the Licensee is acceptable and, therefore, is approved. The safety evaluation is also available at the public document rooms previously listed.

On the basis of the NRC staff's review of various plant-specific and industry programs implemented by the Licensee, the NRC staff concluded that the Licensee took appropriate actions to address embrittlement and cracking in, and thus to ensure the reliability of, the OCNGS reactor vessel internal components.

Based on the above, the staff has concluded that suspension of the Oyster Creek Nuclear Generating Station operating license due to embrittlement and cracking of the reactor vessel internal components is not warranted. As stated previously, continued monitoring of reactor vessel internals as required by the ASME Code and the licensee's inservice inspection program will provide assurance that degradation of components will be detected and appropriate action will be taken.

B. Petitioners request that the NRC suspend the OCNGS operating license until the Licensee provides an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components. The majority of reactor internals are fabricated from high-toughness materials such as stainless steel and were designed with significant margins on allowable stresses. As such, cracking must be severe to adversely impact plant safety. It is unlikely that licensee inspections would not find such severe degradation. In fact, identification and sizing of the cracks in the H4 location on the OCNGS core shroud are good examples of the effectiveness of the inspections. In addition, NRC staff evaluation of the results from internals inspections performed to date at OCNGS resulted in the conclusion that ASME Code safety margins have been maintained.

The Licensee has not provided an analysis to NRC that addresses the synergistic effects of cracking in multiple safety-class components. The NRC staff does not consider the lack of such an analysis to be a safety concern because of the inspection requirements that pertain to reactor internals and the results of inspections performed to date. See Section III.A, *supra.*

Continued monitoring of reactor vessel internals as required by the ASME Code and the licensee's inservice inspection program will provide information about the structural integrity of reactor vessel internals in the long term. The NRC has asked the BWR Vessel Internals Project (BWRVIP), an industry group, to develop an assessment to address cracking in BWR reactor vessel internals. A report from the BWRVIP is expected on the long term effects of reactor vessel internals cracking in late 1995. In addition, the NRC has undertaken a longer term evaluation of the effects of cracking in multiple reactor vessel internal components that will be approached with appropriate treatment of the key variables (safety function, material susceptibility, loading, environment, etc.).

Based on the above, the staff has concluded that suspension of the Oyster Creek Nuclear Generating Station license, due to the lack of an analysis of the synergistic effects of through-wall cracking of safety-class reactor internal components, is not warranted.

IV. Conclusion

The Petitioners requested that the NRC suspend the operating license of Oyster Creek Nuclear Generating Station until: (1) the Licensee inspects, repairs, or replaces, all safety-class reactor internal components subject to embrittlement and cracking, and (2) the Licensee provides an analysis regarding the synergistic effects of through-wall cracking of multiple safety-class components. For the reasons discussed above, I conclude that the issues raised by the Petitioners are being adequately addressed and that there is no basis for suspending the OCNGS operating license or taking the other requested action. Accordingly, the Petitioners' above-referenced requests are denied.

A copy of this Partial Director's Decision will be filed with the Secretary of the Commission for review as stated in 10 CFR 2.206(c). This Decision will become the final action of the Commission 25 days after issuance unless the Commission, on its own motion, institutes review of the Decision within that time.

Dated at Rockville, Maryland, this 4th day of August 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 95–19766 Filed 8–9–95; 8:45 am]

BILLING CODE 7590–01–P

[Docket Nos. 50–280 and 50–281]

Virginia Electric and Power Co.; Issuance of Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 203 and 203 to Facility Operating License Nos. DPR–32 and DPR–37 issued to Virginia Electric and Power Company, which revised the License and the Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2 located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments modified the Licenses and the Technical Specifications to increase the authorized core power level for Surry, Units 1 and 2, from 2441 MWt to 2546 MWt.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the **Federal Register** on December 16, 1994 (59 FR 65085). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement.

Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (60 FR 32356).

For further details with respect to the action see (1) the application for amendment dated August 30, 1994, and supplemented February 6, February 13, February 27, March 23, March 28, April 13, April 20, April 28, May 5, and June 8, 1995, (2) Amendment Nos. 203 and 203 to License Nos. DPR–32 and DPR–37, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items 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 Swen Library, College of William and Mary, Williamsburg, Virginia 23185.

Dated at Rockville, Maryland, this 3rd day of August 1995.

For The Nuclear Regulatory Commission.

Bart C. Buckley,

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

[FR Doc. 95–19767 Filed 8–9–95; 8:45 am]

BILLING CODE 7590–01–P

SECURITIES AND EXCHANGE COMMISSION

Requests Under Review by Office of Management and Budget

Agency Clearance Officer: Michael E. Bartell (202) 942–8800