Dated: September 26, 1997.

**Bea Hardesty,**

*Federal Register Liaison Officer.*

[FR Doc. 97–26056 Filed 9–26–97; 2:57 pm]

**BILLING CODE 7533–01–P**

## NUCLEAR REGULATORY COMMISSION

**[Docket No. 50–302]**

## Florida Power 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. DPR 72, issued to the Florida Power Corporation, (FPC or the licensee), for operation of the Crystal River Nuclear generating Unit 3 (CR3) located in Citrus County, Florida.

The proposed amendment involves a revision to the Emergency Diesel Generator (EDG) protective relaying scheme at CR3, as described in the Final Safety Analysis Report (FSAR) Chapter 8. FPC has evaluated the proposed modifications pursuant to 10 CFR 50.59 and has determined that these modifications constitute an unreviewed safety question (USQ) based on a resulting increase in the probability of a malfunction of equipment important to safety. Therefore, FPC is requesting amendment of the CR3 license to resolve that USQ. The proposed modification will add new protective relays to each EDG generator output breaker to provide additional protection for a potential electrical fault or overpower condition.

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. Does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The EDGs perform a support function for Design Basis Accident mitigation by providing a source of emergency AC electrical power for the Engineered Safeguards loads. For most Design Basis Accidents, a coincident Loss-of-Offsite-Power is postulated to occur and any single random electrical failure is considered credible including complete failure for one EDG to energize the associated 4160V ES bus. The failure of an EDG to energize the associated 4160V ES bus is not a precursor for any postulated Design Basis Accident except Station Blackout (SBO). The failure of both EDGs concurrent with a Loss-of-Offsite-Power causes a Station Blackout. Therefore, any increase in the probability that an EDG will not energize the associated 4160V ES bus will increase the probability of a Station Blackout.

The new relaying added to each EDG has a small probability of spuriously actuating, resulting in a small increase in the probability of an EDG failing to energize the associated 4160V ES bus. Spurious actuation of the overcurrent relaying for the load carrying 4160V ES bus offsite power source breaker will cause a loss of power on the 4160V ES bus and prevent the EDG from re-energizing the bus. In addition, a spurious actuation of the device-32X directional power auxiliary relay can cause a loss of offsite power for the associated 4160V ES bus. This spurious actuation also increases the probability of a Station Blackout. The only new system interfaces are between the EDG and 4160V ES bus systems. The modified relaying will not directly affect the fuel cladding, the Reactor Coolant System (RCS) pressure boundary, or the containment building.

The increase in the probability of a Station Blackout is negligible. Although EDG availability is a contributor to the risk of Station Blackout, the CR–3 licensing basis assumes this event without regard to EDG reliability. Therefore, the probability of previously evaluated accidents is not significantly increased. The new protective relaying could shorten the duration of an actual Station Blackout if a 4160V ES bus fault or other similar problem was a contributor to the event by limiting the damage to the station power systems.

The modified relaying will not increase the consequences of a Station Blackout since both EDGs and offsite power are assumed to be unavailable. The new protective relaying will not create any new timing or sequencing impact to the ES loads supplied from the 4160V ES bus. The small increase in probability that an EDG will not energize the associated 4160V ES bus does not invalidate the Design Basis Accident assumption that one EDG successfully energizes the associated 4160V ES bus (single failure proof). Therefore, the conclusions concerning fission product releases in the FSAR will not be changed.

2. Does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The modified relaying will not directly affect the fuel cladding, the Reactor Coolant System (RCS) pressure boundary, or the containment building. The modifications only impact the EDGs and 4160V ES buses.

The failure of one of the EDGs to energize the associated 4160V ES bus during a Design Basis Accident is a standard "single failure" for determining the acceptability of an accident mitigation system. A standby EDG and the associated 4160V ES bus are not capable of creating an accident such as a Loss-of-Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

There is a small increase in the probability that an EDG will not successfully energize the associated 4160V ES bus. However, the Design Basis Accident assumption that one EDG does successfully energize the bus remains valid. Therefore, no new accident involving the failure of both EDGs other than a Station Blackout needs to be postulated. The proposed modifications to the EDG relaying and the small increase in the probability that an EDG will not energize the associated 4160V ES bus do not introduce any new interfaces or mechanisms that could challenge any fluid system or fission product barrier in a different way than previously evaluated. Therefore, the modifications cannot create the possibility of an accident of a different type than previously evaluated in the FSAR.

3. Does not involve a significant reduction in the margin of safety.

The Bases of the CR–3 technical specifications do not identify a "margin of safety" for the EDGs or 4160V ES buses that is applicable to the proposed EDG relaying modifications. Therefore, the plant response to Design Basis Accidents was evaluated. The accident analysis assumptions remain valid with the existing and proposed changes to the EDG and 4160V ES bus protective relaying. Plant response will remain as evaluated in the accident analysis and the calculated primary and secondary pressures and temperatures during evaluated accidents will not be increased by the changes. The reliability of each EDG and associated 4160V ES bus is being insignificantly reduced in order to increase the availability of the EDG and associated 4160V ES bus after a fault or overcurrent condition occurs. A spurious actuation of one of the added relays might cause one EDG to fail to energize one 4160V ES bus but would not result in failure of the other EDG to perform its function. Therefore, the changes do not reduce the margin of safety in the bases for any Improved Technical Specification.

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. 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 Review and Directives Branch, Division of Freedom of Information and Publications 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 6D22, 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 October 30, 1997, 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 Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

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 a 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 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 R. Alexander Glenn, General Counsel, Florida Power Corporation, MAC—A5A, P. O. Box 14042, St. Petersburg, Florida 33733–4042, 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 September 12, 1997, 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 Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 34428.

Dated at Rockville, Maryland, this 23rd day of September 1997.

For The Nuclear Regulatory Commission.

**L. Raghavan, Sr.,**

*Project Manager, Project Directorate II–3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.*

[FR Doc. 97–25899 Filed 9–29–97; 8:45 am]

BILLING CODE 2590–01–P

## NUCLEAR REGULATORY COMMISSION

**[Docket Nos. 50–338]**

## In the Matter of Virginia Electric and Power Company North Anna Power Station, Unit Nos. 1 and 2; Exemption and 50–339

### I

The Virginia Electric and Power Company (the licensee) is the holder of Facility Operating License Nos. NPF–4 and NPF–7, which authorize operation of the North Anna Power Station (NAPS), Unit Nos. 1 and 2. The licenses provide, among other things, that the licensee be subject to all rules, regulations, and Orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

The facility consists of two pressurized-water reactors at the licensee's site located in Louisa County, Virginia.

### II

The Code of Federal Regulations at 10 CFR 70.24, ''Criticality Accident Requirements,'' requires that each licensee authorized to possess special nuclear material (SNM) shall maintain a criticality accident monitoring system in each area where such material is handled, used, or stored. Subsections (a)(1) and (a)(2) of 10 CFR 70.24 specify detection, sensitivity and coverage capabilities of the monitors required by 10 CFR 70.24(a). Subsection (a)(3) requires licensees to maintain emergency procedures for each area in which this licensed SNM is handled, used, or stored.

Subsection (d) of 10 CFR 70.24 states that any licensee who believes that there is good cause why it should be granted an exemption from all or part of 10 CFR 70.24 may apply to the Commission for such an exemption and shall specify the reasons for the relief requested.

### III

By letter dated January 28, 1997, as supplemented March 24, 1997, Virginia Electric and Power Company requested an exemption from 10 CFR 70.24(a). The Commission technical staff has reviewed the licensee's submittal and has determined that inadvertent criticality is not likely to occur in SNM handling or storage areas at NAPS, Units 1 and 2.

At North Anna, SNM is present principally as nuclear fuel. Other small quantities of SNM are used on site. However, the total amount used in non-fuel applications is significantly less than the quantity specified in 10 CFR 70.24(a). The small quantity of non-fuel SNM present, and the form in which it is stored and used, precludes an inadvertent criticality. Therefore, SNM used as nuclear fuel is the only material on site subject to the requirements of 10 CFR 70.24(a).

Nuclear fuel is stored in the new fuel storage area and the spent fuel pool. New fuel is stored dry (in air) in the new fuel storage area. The spent fuel pool is used to store irradiated fuel under water after its discharge from the reactor, and new fuel prior to loading into the reactor.

The new fuel storage area is used to receive and store new fuel in a dry condition upon arrival on site and prior to loading in the reactor or spent fuel pool. The spacing between new fuel assemblies in the storage racks is sufficient to maintain the array in a subcritical condition even under accident conditions assuming the presence of moderator. The maximum nominal enrichment of 4.3 wt% U–235 for the new fuel assemblies results in a maximum $k_{eff}$ of less than 0.95 under conditions of accidental flooding by unborated water and $k_{eff}$ less than 0.98 under conditions of low-density optimum moderation. The staff has found the design of the licensee's new fuel storage racks to be adequate to store fuel enriched to 4.3 wt% U–235.

Consistent with Technical Specification Section 5.6.1.1, the spent fuel pool is designed to store the fuel in a geometric array that precludes criticality. The spent fuel racks are designed such that the effective neutron multiplication factor, $k_{eff}$, will remain less than or equal to 0.95 under all normal and accident conditions for fuel of maximum nominal enrichment of 4.3 wt% U–235.

Nuclear fuel is moved between the shipping container, the new fuel storage racks, the reactor vessel, and the spent fuel pool to accommodate refueling operations. In all cases, fuel movements are procedurally controlled and designed to preclude conditions involving criticality concerns.

The purpose of the criticality monitors required by 10 CFR 70.24 is to ensure that if a criticality were to occur during the handling of nuclear material, personnel would be alerted to that fact and would take appropriate action. Although the staff has determined that such an accident is not likely to occur, the licensee has radiation monitors, as required by General Design Criterion 63, in fuel storage and handling areas. These monitors have associated area alarms and control room annunciators and would detect excessive radiation levels and will alert personnel to allow them to initiate appropriate emergency procedures and safety actions. The low probability of an inadvertent criticality together with the licensee's adherence to General Design Criterion 63 constitute good cause for granting an exemption to the requirements of 10 CFR 70.24(a).

### IV

The Commission has determined that, pursuant to 10 CFR 70.14, this exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest; therefore, the Commission hereby grants Virginia Electric and Power Company the exemption from the requirements of 10 CFR 70.24(a) for North Anna Power Station, Unit Nos. 1 and 2, relating to criticality accident monitoring requirements.

### V

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not result in any significant adverse environmental impact (62 FR 49540).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 23rd day of September 1997.

For The Nuclear Regulatory Commission.

**Frank J. Miraglia,**

*Acting Director, Office of Nuclear Reactor Regulation.*

[FR Doc. 97–25900 Filed 9–29–97; 8:45 am]

BILLING CODE 7590–01–P