hearing will be published in the **Federal Register** and served on the parties to the hearing.

For further details with respect to this Order, see the application for approval filed by NMPC under cover of a letter dated July 21, 1998, from John H. Mueller of NMPC, as supplemented by letter dated October 23, 1998, and the safety evaluation dated December 11, 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 Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Dated at Rockville, Maryland, this 11th day of December 1998.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 98–33587 Filed 12–17–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–286]

Power Authority of the State of New York (Indian Point Nuclear Generating Unit No. 3); Exemption

I

The Power Authority of the State of New York (the licensee) is the holder of Facility Operating License No. DPR–64, which authorizes operation of the Indian Point Nuclear Generating Unit No. 3 (IP3). The license provides that the licensee is subject to all rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

The facility consists of a pressurized-water reactor at the licensee's site located in Westchester County, New York.

II

The Code of Federal Regulations, 10 CFR 70.24, ''Criticality Accident Requirements,'' requires that each licensee authorized to possess special nuclear material shall maintain a criticality accident monitoring system in each area where such material is handled, used, or stored. Subsection (a)(1) and (a)(2) of 10 CFR 70.24 specifies detection and sensitivity requirements that these monitors must meet. Subsection a(1) also specifies that all areas subject to criticality accident monitoring must be covered by two detectors. Subsection (a)(3) of 10 CFR 70.24 requires licensees to maintain emergency procedures for each area in which this licensed special nuclear material is handled, used, or stored and provides (1) that the procedures ensure that all personnel withdraw to an area of safety upon the sounding of a criticality accident monitor alarm, (2) that the procedures must include drills to familiarize personnel with the evacuation plan, and (3) that the procedures designate responsible individuals for determining the cause of the alarm and placement of radiation survey instruments in accessible locations for use in such an emergency. Subsection (b)(1) of 10 CFR 70.24 requires licensees to have a means to identify quickly personnel who have received a dose of 10 rads or more. Subsection (b)(2) of 10 CFR 70.24 requires licensees to maintain personnel decontamination facilities, to maintain arrangements for a physician and other medical personnel qualified to handle radiation emergencies, and to maintain arrangements for the transportation of contaminated individuals to treatment facilities outside the site boundary. Paragraph (c) of 10 CFR 70.24 exempts Part 50 licensees from the requirements of paragraph (b) of 10 CFR 70.24 for special nuclear material used or to be used in the reactor. Subsection (d) of 10 CFR 70.24 states that any licensee who believes that there is good cause why he 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

The special nuclear material that could be assembled into a critical mass at IP3 is in the form of nuclear fuel; the quantity of special nuclear material other than fuel that is stored on site is small enough to preclude achieving a critical mass. The Commission technical staff has evaluated the possibility of an inadvertent criticality of the nuclear fuel at IP3 and has determined that such an accident cannot occur if the licensee meets the following seven criteria:

1. Plant procedures permit only one new fuel assembly to be in transit between the associated shipping cask and dry storage rack.
2. The k-effective does not exceed 0.95, at a 95% probability, 95% confidence level in the event that the fresh fuel storage racks are filled with fuel of the maximum permissible U-235 enrichment and flooded with pure water.
3. If optimum moderation of fuel in the fresh fuel storage racks occurs when the fresh fuel storage racks are not flooded, the k-effective corresponding to this optimum moderation does not exceed .98, at a 95 percent probability, 95 percent confidence level.
4. The k-effective does not exceed 0.95, at a 95% probability, 95% confidence level in the event that the spent fuel storage racks are filled with fuel of the maximum permissible U-235 enrichment and flooded with pure water.
5. The quantity of forms of special nuclear material, other than nuclear fuel, that are stored on site in any given area is less than the quantity necessary for a critical mass.
6. Radiation monitors are provided in fuel storage and handling areas to detect excessive radiation levels and to initiate appropriate safety actions.
7. The maximum nominal U–235 enrichment is limited to 5 wt%.

By letter dated September 24, 1998, the licensee requested an exemption from 10 CFR 70.24. In this exemption request, the licensee addressed the seven criteria given above. The Commission's technical staff has reviewed the licensee's submittal and has determined that IP3 meets the criteria for prevention of inadvertent criticality; therefore, the staff has determined that there is no credible way in which an inadvertent criticality could occur in special nuclear materials handling or storage areas at IP3.

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 special nuclear material personnel would be alerted to that fact and would take appropriate action. The staff has determined that there is no credible way in which such an accident could occur; furthermore, the licensee has radiation monitors, as required by General Design Criterion (GDC) 63, in fuel storage and handling areas. These monitors will alert personnel to excessive radiation levels and allow them to initiate appropriate safety actions. The low probability of an inadvertent criticality together with the licensee's adherence to GDC 63 constitute good cause for granting an exemption to the requirements of 10 CFR 70.24.

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 the following exemption:

The Power Authority of the State of New York is exempt from the requirements of 10 CFR 70.24 for Indian Point Unit No. 3.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will have no significant impact on the quality of the human environment [63 FR 68315].

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 10th day of December 1998.

For the Nuclear Regulatory Commission.

Samuel J. Collins,

Director, Office of Nuclear Reactor Regulation.

[FR Doc. 98–33586 Filed 12–17–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–271]

Vermont Yankee Nuclear 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–28 issued to Vermont Yankee Nuclear Power Corporation (the licensee) for operation of the Vermont Yankee Nuclear Power Station located in Windham County, Vermont.

The proposed amendment would allow intermittent opening of manual primary containment isolation valves with appropriate administrative controls. Opening these valves is necessary to perform routine evolutions such as surveillances, sampling and venting/draining of plant systems.

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 the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change allows an isolated primary containment penetration to be opened as necessary to meet operational objectives defined in applicable Technical Specifications and/or approved plant procedures. Primary containment isolation is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. Although primary containment isolation is considered in the mitigation of the consequences of an accident, administrative controls provide acceptable compensatory actions to assure the penetration is isolated in the event of an accident. Therefore, the consequences of a previously analyzed event that may occur during the opening of the isolated line are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any previously evaluated?

This change allows temporary breaches of the primary containment boundary under strict administrative controls, for the purposes of conducting normal operational evolutions required by other Technical Specifications and/or approved plant procedures. In the event containment isolation is required while any flow path is open under administrative controls, provisions exist to isolate that flow path with a single active-failure-proof boundary as required by the primary containment Technical Specification Limiting Conditions for Operation. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does the change involve a significant reduction in a margin of safety?

The margin of safety considered in determining the required compensatory action is also based on providing the single active-failure-proof boundary. Opening of primary containment penetrations on an intermittent basis is required for performance of routine evolutions as noted previously. Plant procedures administratively control the opening and closing of the affected valves. The administrative controls are defined in the Technical Specifications Bases. When a manual valve is opened under these conditions, a dedicated operator, with whom Control Room communication is immediately available, is stationed in the immediate vicinity of the valve controls. In the event primary containment must be rapidly reinstated, this individual will close the valve in an expeditious manner. Once closed, this flow path will meet the same single active-failure-proof criteria as other containment penetrations. Since the flow path will be closed promptly on a containment isolation demand, the valve will be open only slightly longer than if it had been closed by an automatic actuator. Therefore, this change does not involve a significant reduction in a margin of safety.

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