[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 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 at subsection (a) of 10 CFR 70.24, ''Criticality Accident Requirements,'' requires that each licensee authorized to possess special nuclear material shall maintain in each area where such material is handled, used, or stored, a criticality monitoring system ''using gamma- or neutron-sensitive radiation detectors which will energize clearly audible alarm signals if accidental criticality occurs.'' Subsection (a)(1) of 10 CFR 70.24 specifies the detection, sensitivity, and coverage capabilities of the monitors required by 10 CFR 70.24(a). The specific requirements of subsection (a)(1) are that ''the monitoring system shall be capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute.'' Subsection (a)(3) of 10 CFR 70.24 requires that the licensee shall 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 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 (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.

The purpose of 10 CFR 70.24 (a), (a)(1), and (a)(3) is to ensure that any inadvertent criticality is detected and that action is taken to protect personnel and correct the problem. By letter dated December 20, 1996, as supplemented March 5, 1997, and March 19, 1997, the licensee requested an exemption from the requirements of 10 CFR 70.24. The licensee proposes to handle and store unirradiated fuel without having the criticality monitoring system specified in 10 CFR 70.24. The licensee also proposes to handle and store unirradiated fuel without the speicfic emergency procedures detailed in 10 CFR 70.24. The licensee believes that fuel handling procedures and design features make an inadvertent criticality unlikely. The licensee believes that a portable radiation monitoring system and existing plant procedures will provide adequate protection in the unlikely event of an accidental criticality. The licensee also believes that current emergency procedures and training are adequate to meet the intent of 10 CFR 70.24(a)(3).

III

Special nuclear material, as nuclear fuel, is stored in the spent fuel pool or the new (unirradiated) fuel storage racks. 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 racks are used to store new fuel in a dry condition upon arrival on site.

Special nuclear material is also present in the form of fissile material incorporated into fission chambers for nuclear instrumentation, primary source assemblies, and Health Physics calibration sources. The small quantity of special nuclear material present in these items precludes an inadvertent criticality.

Consistent with Technical Specification Section 5.4, the spent fuel pool is designed to store the fuel in a geometric array using a solid neutron absorber 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 normal and accident conditions for fuel of maximum enrichment of 5.0 wt% U–235. The staff has found this design adequate.

The new fuel storage racks may be 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 enrichment of 5.0 wt% U–235 for the new fuel assemblies results in a maximum K_{eff} of less than 0.95 under conditions of accidental flooding. The staff has found the design of the licensee's new fuel storage racks to be adequate to store fuel enriched to no greater than 5.0 wt% U–235.

Nuclear fuel is moved between the new fuel storage racks, the reactor vessel, and the spent fuel pool to accommodate refueling operations. In addition, fuel is moved into the facility and within the reactor vessel, or within the spent fuel pool. Fuel movements are procedurally controlled and designed to preclude conditions involving criticality concerns. Fuel handling procedures and the design features of the fuel handling system are discussed in the licensee's Final Safety Analysis Report.

Technical Specification Section 3.8 precludes certain movements of heavy loads over the spent fuel pool to prevent a fuel handling accident. Previous accident analyses have demonstrated that a fuel handling accident (i.e., a dropped fuel assembly) will not create conditions which could result in inadvertent criticality.

Procedures and controls prevent an inadvertent criticality during fuel handling; nevertheless the licensee will provide monitoring in the IP3 Fuel Storage Building during dry fuel handling operations. During dry fuel handling operations, the licensee will have in operation at least one portable detector that will meet the detection and sensitivity criteria of Sections 5.6 and 5.7 of ANSI/ANS 8.3 (1986), ''American National Standard Criticality Accident Alarm System.'' Upon detection, this instrument shall automatically cause an immediate alarm audible in all areas from which evacuation is necessary to minimize exposure. The staff has determined that the detection and sensitivity criteria in the ANSI standard are as rigorous as those specified in 10 CFR 70.24(a)(1). The staff has also determined that, because fuel handling equipment design and procedures make a criticality unlikely, one detector will be adequate and that in the case of fuel handling at IP3 two detectors as required by 10 CFR 70.24(a)(1) are not necessary.

The licensee has procedures and conducts training on dealing with radiological emergencies consistent with 10 CFR 50.47 and Part 50, Appendix E. In addition to this training, the licensee gives training on responding to a criticality monitor alarm to radiation workers accessing the fuel handling building. This training will be provided as necessary until dry fuel handling in 1997 is complete and the subject material has been incorporated into general employee training. The staff has determined that the licensee's procedures and training meet the intent of 10 CFR 70.24(a)(3); therefore, adherence to the specific requirements of this section is not necessary to serve the underlying purpose of the rule.

Because inadvertent criticality is precluded by both design and procedure, because adequate radiation monitoring is present, and because the licensee maintains emergency procedures for the areas in which fuel is handled, the staff has concluded that there is reasonable assurance that irradiated and unirradiated fuel will remain subcritical; furthermore, there is reasonable assurance that, should an inadvertent criticality occur, the licensee will detect such a criticality and workers will respond properly. The combination of plant design features, fuel handling procedures, the use of a portable criticality monitor, radiological emergency procedures and radiation worker training constitute good cause for granting an exemption to the requirements of 10 CFR 70.24.

IV

Accordingly, 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(a), 10 CFR 70.24(a)(1), and 10 CFR 70.24(a)(3) for Indian Point Nuclear Generating Unit No. 3. This exemption is contingent on the facility's maintaining the hardware, procedure, and training described in Section III above.

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 (62 FR 14705).

This exemption is effective upon issuance.

Dated at Rockville, MD, this 27th day of March 1997.

For the Nuclear Regulatory Commission,

Frank J. Miraglia, Jr.,

Acting Director, Office of Nuclear Reactor Regulation.

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[Docket 70–7001]

Notice of Amendment to Certificate of Compliance GDP–1 for the U.S. Enrichment Corporation, Paducah Gaseous Diffusion Plant, Paducah, Kentucky

The Director, Office of Nuclear Material Safety and Safeguards, has made a determination that the following amendment request is not significant in accordance with 10 CFR 76.45. In making that determination the staff concluded that (1) there is no change in the types or significant increase in the amounts of any effluents that may be released offsite; (2) there is no significant increase in individual or cumulative occupational radiation exposure; (3) there is no significant construction impact; (4) there is no significant increase in the potential for, or radiological or chemical consequences from, previously analyzed accidents; (5) the proposed changes do not result in the possibility of a new or different kind of accident; (6) there is no significant reduction in any margin of safety; and (7) the proposed changes will not result in an overall decrease in the effectiveness of the plant's safety, safeguards or security programs. The basis for this determination for the amendment request is shown below.

The NRC staff has reviewed the certificate amendment application and concluded that it provides reasonable assurance of adequate safety, safeguards, and security, and compliance with NRC requirements. Therefore, the Director, Office of Nuclear Material Safety and Safeguards, is prepared to issue an amendment to the Certificate of Compliance for the Paducah Gaseous Diffusion Plant. The staff has prepared a Compliance Evaluation Report which provides details of the staff's evaluation.

The NRC staff has determined that this amendment satisfies the criteria for a categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for this amendment.

USEC or any person whose interest may be affected may file a petition, not exceeding 30 pages, requesting review of the Director's Decision. The petition must be filed with the Commission not later than 15 days after publication of this **Federal Register** Notice. A petition for review of the Director's Decision shall set forth with particularity the interest of the petitioner and how that interest may be affected by the results of the decision. The petition should specifically explain the reasons why review of the Decision should be permitted with particular reference to the following factors: (1) The interest of the petitioner; (2) how that interest may be affected by the Decision, including the reasons why the petitioner should be permitted a review of the Decision; and (3) the petitioner's areas of concern about the activity that is the subject matter of the Decision. Any person described in this paragraph (USEC or any person who filed a petition) may file a response to any petition for review, not to exceed 30 pages, within 10 days after filing of the petition. If no petition is received within the designated 15-day period, the Director will issue the final amendment to the Certificate of Compliance without further delay. If a petition for review is received, the decision on the amendment application will become final in 60 days, unless the Commission grants the petition for review or otherwise acts within 60 days after publication of this **Federal Register** Notice.

A petition for review must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW, Washington, DC, by the above date.

For further details with respect to the action see (1) the application for amendment and (2) the Commission's Compliance Evaluation Report. 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.

Date of amendment request: February 14, 1997, revised March 10, 1997.

Brief description of amendment: The amendment revises the Technical Safety Requirement for the design features for the cranes in the feed facilities and reflects the associated changes to the Safety Analysis Report.

Basis for finding of no significance:

1. The proposed amendment will not result in a change in the types or significant increase in the amounts of any effluents that may be released offsite.

The proposed change to TSR 2.2.5.2 involves a change to the design features of the hoist brakes for the feed facility cranes. These changes have no impact on plant effluents and will not result in any impact to the environment.

2. The proposed amendment will not result in a significant increase in individual or cumulative occupational radiation exposure.