NUCLEAR REGULATORY COMMISSION

[Docket Nos. STN 50–454, STN 50–455 and STN 50–456, STN 50–457]

Commonwealth Edison Company; (Byron Station, Units 1 and 2); (Braidwood Station, Units 1 and 2); Exemption

I

Commonwealth Edison Company (ComEd, the licensee) is the holder of Facility Operating License Nos. NPF–37, NPF–66, NPF–72, and NPF–77, which authorize operation of Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, respectively. The licenses provide, among other things, that the licensee is subject to all rules, regulations, and orders of the Commission now or hereafter in effect.

The Byron facility consists of two pressurized-water reactors located at the licensee's site in Ogle County, Illinois. The Braidwood facility consists of two pressurized-water reactors located at the licensee's site in Will County, Illinois.

II

In its letter dated April 3, 1997, as supplemented on June 19, 1997, ComEd requested an exemption from the Commission's regulations. Title 10 of the Code of Federal Regulations, Part 50, Section 60 (10 CFR 50.60), ''Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation,'' states that all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary as stated in Appendices G and H to 10 CFR Part 50. Appendix G to 10 CFR Part 50 defines pressure-temperature (P-T) limits during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime, and specifies that these P-T limits must be at least as conservative as the limits obtained by conforming to the methods of analysis and the margins of safety of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G. 10 CFR 50.55a requires that any reference to ASME Code Section XI in 10 CFR Part 50 refers to addenda through the 1988 Addenda and editions through the 1989 Edition of the Code unless otherwise noted. It is specified in 10 CFR 50.60(b) that alternatives to the requirements described in Appendices G and H to 10 CFR Part 50 may be used when an exemption is granted by the Commission under 10 CFR 50.12.

To mitigate low-temperature overpressure transients that would produce pressure excursions exceeding the required limits while the reactor is operating at low temperatures, the licensee installed a low-temperature overpressure protection (LTOP) system. The system includes pressure-relieving devices called power-operated relief valves (PORVs). The PORVs are set at a pressure low enough so that if an LTOP transient occurred, the mitigation system would prevent the pressure in the reactor vessel from exceeding the required limits. To prevent the PORVs from lifting as a result of normal operating pressure surges, some margin is needed between the PORV setpoint and the normal operating pressure. In addition, when instrument uncertainty is considered, the operating window between the PORV setpoint and the minimum pressure required for reactor coolant pump seals is small and presents difficulties for plant operation.

The licensee has requested the use of the 1996 Addenda to the ASME Code, Section XI, Appendix G, which allows the use of lower stress intensity factors for determining the applied stress intensity from pressure and thermal stresses, and allows use of an LTOP system setpoint so that system pressure does not exceed 110 percent of the P-T limits. The 1996 Addenda to the ASME Code, Section XI, Appendix G, is consistent with guidelines developed by the ASME Working Group on Operating Plant Criteria to define pressure limits during LTOP events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of pressure-relieving devices used for LTOP. ASME Code, Section XI, Appendix G, 1996 Addenda, has been approved by the ASME Code Committee.

III

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested entity or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 50 when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security; and (2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), ''Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. * * *''

The underlying purpose of 10 CFR 50.60 and 10 CFR Part 50 Appendix G is to establish fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences, to which the pressure boundary may be subjected over its service lifetime. Section IV.A.2 of Appendix G to 10 CFR Part 50, requires that the reactor vessel be operated with P–T limits at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of Section XI of the ASME Code. 10 CFR 50.55a requires that any reference to ASME Code Section XI in 10 CFR Part 50, Appendix G, refers to addenda through the 1988 Addenda and editions through the 1989 Edition of the ASME Code, unless otherwise noted.

Appendix G of the ASME Code requires that the P-T limits be calculated: (a) Using a safety factor of two on the principal membrane (pressure) stresses, (b) assuming a flaw at the surface with a depth of one-quarter of the vessel wall thickness (¼ T) and a length of six (6) times its depth, and (c) using a conservative fracture toughness curve that is based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the reactor vessel material.

For determining the P–T limits, the licensee proposed to use the safety margins based on the 1996 Addenda to the ASME Code in lieu of the 1989 Edition. When compared to the 1989 Edition of the ASME Code, the 1996 Addenda permits the use of a lower stress intensity factor for determining the applied stress intensity from pressure and thermal stresses. This results in a slight reduction in the applied stress intensity and a corresponding shift in the allowable pressure at a given temperature in the non-conservative direction; however, this difference is minor when compared to the explicit conservatisms incorporated into Appendix G, and the changes in the stress intensity factor are supported by the work performed for NRC and for others by J.A. Keeney and T.L. Dickson at Oak Ridge National Laboratory (ORNL).

For determining the LTOP system setpoint, the licensee proposed to use safety margins based on the 1996 Addenda to the ASME Code. The 1996

Addenda allows determination of the setpoint for mitigating LTOP events so that the maximum pressure in the vessel would not exceed 110 percent of the P–T limits that are determined using the 1996 methodology. This results in a safety factor of 1.8 on the principal membrane stresses. All other factors, including assumed flaw size and fracture toughness, remain the same. Although this methodology would reduce the safety factor on the principal membrane stresses, the proposed criteria will provide adequate margins of safety for the reactor vessel during LTOP transients and, thus, will satisfy the underlying purpose of 10 CFR 50.60 for fracture toughness requirements. Further, by relieving the operational restrictions, the potential for undesirable lifting of the PORV would be reduced, thereby improving plant safety.

It should be noted that the provision to set the PORV setpoint so that system pressure remains below 110 percent of the P–T limits has already been incorporated into the Byron and Braidwood licensing basis. This provision was approved by an exemption to 10 CFR 50.60 granted to Byron, Units 1 and 2, on November 29, 1996, to Braidwood, Unit 1 on July 13, 1995, and to Braidwood, Unit 2 on December 12, 1997, to allow the use of ASME Code Case N–514. Therefore, although it represents a change from the 1989 Edition of the ASME Code, it is not a change to the current licensing basis for the facilities.

IV

For the foregoing reasons, the NRC staff has concluded that ComEd's proposed use of the alternate methodology in determining the acceptable setpoint for LTOP events will not present an undue risk to public health and safety and is consistent with the common defense and security. The NRC staff has determined that there are special circumstances present, as specified in 10 CFR 50.12(a)(2), in that 10 CFR 50.60 need not be applied in order to achieve the underlying purpose of this regulation, which is to provide adequate fracture toughness of the reactor pressure boundary.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), an exemption is authorized by law, will not endanger life or property or common defense and security, and is, otherwise, in the public interest. Therefore, the Commission hereby grants an exemption from the requirements of 10 CFR 50.60 so that the P–T limits may be determined using the 1996 Addenda to the ASME Code, Section XI, Appendix G, and the LTOP system setpoint may be determined so that system pressure does not exceed 110 percent of the P–T limits.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not have a significant effect on the quality of the human environment (63 FR 2268).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 16th day of January, 1998.

For the Nuclear Regulatory Commission.

Frank J. Miraglia,

Acting Director, Office of Nuclear Reactor Regulation.

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–271]

Vermont Yankee Nuclear Power Corporation, Vermont Yankee Nuclear Power Station; Exemption

I

The Vermont Yankee Nuclear Power Corporation (the licensee) is the holder of Facility Operating License No. DPR–28, which authorizes operation of the Vermont Yankee Nuclear Power Station. The license provides, among other things, 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 single-unit boiling-water reactor located at the licensee's site in Windham County, Vermont.

II

Section 70.24 of Title 10 of the Code of Federal Regulations (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 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 SNM is handled, used, or stored and also requires that (1) the procedures ensure that all personnel withdraw to an area of safety upon the sounding of a criticality accident monitor alarm, (2) the procedures must include drills to familiarize personnel with the evacuation plan, and (3) 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 for identifying 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 the services of 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 SNM used or to be used in the reactor. Paragraph (d) of 10 CFR 70.24 states that any licensee who believes that there is good cause why he or she 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 SNM that could be assembled into a critical mass at Vermont Yankee is in the form of nuclear fuel; the quantity of SNM other than fuel that is stored on site in any given location is small enough to preclude achieving a critical mass. The Commission's technical staff has evaluated the possibility of an inadvertent criticality of the nuclear fuel at Vermont Yankee and has determined that it is extremely unlikely for such an accident to occur if the licensee meets the following seven criteria:

1. Only three new fuel assemblies are allowed out of a shipping cask or storage rack at one time.

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 occurs at low moderator density, then the k-effective does not exceed 0.98, 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 a