the licensee on October 11, 1994. Enforcement action is pending. NRC is continuing its review.

* * * * *

A copy of NUREG–0090, Vol. 17, No. 4 is available for inspection or copying for a fee at the NRC Public Document Room, 2120 L Street NW., (lower level), Washington, DC 20037, or at any of the nuclear power plant Local Public Document Rooms throughout the country.

Copies of this report (or any of the previous reports in this series), may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013–7082. A year's subscription to the NUREG–0090 series publication, which consists of four issues, is also available.

Copies of the report may also be purchased from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

Dated at Rockville, MD this 3rd day of July 1995.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 95–16808 Filed 7–7–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. STN 50–456]

Commonwealth Edison Company; Braidwood Station, Unit 1; Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an exemption from Facility Operating License No. NPF–72, issued to the Commonwealth Edison Company (the licensee), for Braidwood Station, Unit 1, located in Will County, Illinois.

Environmental Assessment

Identification of Proposed Action

The proposed action requests an exemption from certain requirements of 10 CFR 50.60, ''Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation,'' to allow application of an alternate methodology to determine the low temperature overpressure protection (LTOP) setpoint for Braidwood Station, Unit 1. The proposed alternate methodology is consistent with guidelines developed by the American Society of Mechanical Engineers (ASME) Working Group on Operating Plant Criteria (WGOPC) 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. These guidelines have been incorporated into Code Case N–514, ''Low Temperature Overpressure Protection,'' which has been approved by the ASME Code Committee.

The content of this code case has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. The NRC staff is revising 10 CFR 50.55a, which will endorse the 1993 Addenda and Appendix G of Section XI into the regulations.

The philosophy used to develop Code Case N–514 guidelines is to ensure that the LTOP limits are still below the pressure/temperature (P/T) limits for normal operation, but allow the pressure that may occur with activation of pressure-relieving devices to exceed the P/T limits, provided acceptable margins are maintained during these events. This philosophy protects the pressure vessel from LTOP events, and still maintains the Technical Specification P/T limits applicable for normal heatup and cooldown in accordance with Appendix G to 10 CFR Part 50 and Sections III and XI of the ASME Code. The exemption was requested by the licensee by letter dated November 30, 1994, and supplemented by letter dated May 11, 1995.

The Need for the Proposed Action

In 10 CFR 50.60 it states that all light-water nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary as set forth in Appendices G and H to 10 CFR Part 50. Appendix G to 10 CFR 50 defines 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. It is specified in 10 CFR 50.60(b) that alternatives to the described requirements 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 prevent transients that would produce pressure excursions exceeding the Appendix G P/T limits while the reactor is operating at low temperatures, the licensee installed an LTOP system. The LTOP system includes pressure relieving devices in the form of Power-Operated Relief Valves (PORVs) that are set at a pressure low enough that if a transient occurred while the coolant temperature is below the LTOP enabling temperature, they would prevent the pressure in the reactor vessel from exceeding the Appendix G P/T limits. To prevent these valves from lifting as a result of normal operating pressure surges (e.g., reactor coolant pump starting, and shifting operating charging pumps) with the reactor coolant system in a water solid condition, the operating pressure must be maintained below the PORV setpoint.

In addition, in order to prevent cavitation of a reactor coolant pump, the operator must maintain a differential pressure across the reactor coolant pump seals. Hence, the licensee must operate the plant in a pressure window that is defined as the difference between the minimum required pressure to start a reactor coolant pump and the operating margin to prevent lifting of the PORVs due to normal operating pressure surges. The licensee's LTOP analysis indicates that using the Appendix G safety margins to determine the PORV setpoint would result in a pressure setpoint within its operating window, but there would be no margin for normal operating pressure surges. Therefore, operating with these limits could result in the lifting of the PORVs and cavitation of the reactor coolant pumps during normal operation. Therefore, the licensee proposed that in determining the PORV setpoint for LTOP events for Braidwood, the allowable pressure be determined using the safety margins developed in an alternate methodology in lieu of the safety margins required by Appendix G to 10 CFR Part 50. The alternate methodology is consistent with ASME Code Case N–514.

An exemption from 10 CFR 50.60 is required to use the alternate methodology for calculating the maximum allowable pressure for LTOP considerations.

Environmemntal Impacts of the Proposed Action

The Commission has completed its evaluation of the licensee's application.

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 (1/4) of the vessel wall thickness 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 Braidwood reactor vessel material.

In determining the PORV setpoint for LTOP events, the licensee proposed to use safety margins based on an alternate methodology consistent with the proposed ASME Code N–514 guidelines. The ASME Code Case N–514 allows determination of the setpoint for LTOP events such that the maximum pressure in the vessel would not exceed 110 percent of the P/T limits of the existing ASME Appendix G. 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, use of the proposed criteria will provide adequate margins of safety to the reactor vessel during LTOP transients.

Accordingly, the Commission concludes that this proposed action would result in no significant radiological environmental impact.

With regard to potential non-radiological impacts, the proposed change involves use of more realistic safety margins for determining the PORV setpoint during LTOP events. It does not affect non-radiological plant effluents and has no other environmental impact. Therefore, the Commission concludes that there are no significant non-radiological environmental impacts associated with the proposed exemption.

Alternative to the Proposed Action

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 did not involve the use of any resources not previously considered in the Final Environmental Statements related to operation of Braidwood Station.

Agencies and Persons Consulted

In accordance with its stated policy, on June 15, 1995, the staff consulted with the Illinois State Official, Mr. Frank Niziolek; Head, Reactor Safety Section; Division of Engineering; Illinois Department of Nuclear Safety; regarding the environmental impact of the proposed action. The State official had no comments.

Finding of No Significant Impact

Based upon the foregoing 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 exemption.

For further details with respect to this action, see the request for exemption dated November 30, 1994, as supplemented May 11, 1995, which is available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC and at the local public document room located at the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481.

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

For the Nuclear Regulatory Commission.

Ramin R. Assa,

Project Director, Project Directorate III–2, Division of Reactor Projects–III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–16809 Filed 7–6–95; 8:45 am]

BILLING CODE 7590–01–M

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Thermal Hydraulic Phenomena; Notice of Meeting

The ACRS Subcommittee on Thermal Hydraulic Phenomena will hold a meeting on July 26 and 27, 1995, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

Most of the meeting will be closed to public attendance to discuss Westinghouse Electric Corporation proprietary information pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Wednesday, July 26, 1995—8:30 a.m. until the conclusion of business

Thursday, July 27, 1995—8:30 a.m. until the conclusion of business

The Subcommittee will continue its review of the Westinghouse COBRA/TRAC best-estimate ECCS thermal hydraulic code. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff.

Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, the Westinghouse Electric Corporation, their consultants, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the scheduling of sessions which are open to the public, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting the cognizant ACRS staff engineer, Mr. Paul A. Boehnert (telephone 301/415–8065) between 7:30 a.m. and 4:15 p.m. (edt). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes in the proposed agenda, etc., that may have occurred.

Dated: June 30, 1995.

Sam Duraiswamy,

Chief, Nuclear Reactors Branch.

[FR Doc. 95–16810 Filed 7–7–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket 70–364]

Babcock and Wilcox Company; Parks Township Facility; Director's Decision Under 10 CFR 2.206

Notice is hereby given that the Director, Office of Nuclear Material Safety and Safeguards, has taken action with regards to the remaining issues (Sections Q and X) referred to the Commission's Executive Director for Operations, by the Atomic Safety and Licensing Board, in its Initial Director's Decision, dated January 3, 1995, *Babcock and Wilcox Company* (Pennsylvania Nuclear Service Operation Parks Township, PA), LBP–95–1, 41 NRC 1, 35 (1995). Section Q was interpreted as a request that the NRC test for radioactive contamination in the general vicinity of Kepple Hill and Riverview in Parks Township, and Section X was interpreted as a request