• NRC Inspection Report 50–334/94–81; 50–412/94–81. This document provides the results of an IPAP trial assessment and will be placed in the NRC PDR when issued.

The objectives of the public meeting are to provide a brief description of the Integrated Performance Assessment Process, answer questions on the process, and receive feedback from interested members of the public.

DATES: The meeting will be held on April 11, 1995, from 8:30 a.m. to 12:30 p.m. Persons planning to attend the public information meeting should submit a completed registration form (see below) by April 3, 1995. Interested persons unable to attend the meeting may submit written comments. Submit comments by April 18, 1995. Comments received after this date will be considered if practical to do so, but the Commission is able to assure consideration only for comments received on or before this date.

ADDRESSES: The meeting will be held at the United States Nuclear Regulatory Commission, Two White Flint North, Auditorium, 11545 Rockville Pike, Rockville, MD.

Send completed registration forms to Mr. David L. Gamberoni, M/S OWFN 12–E–4, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Send comments to Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publication Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555–0001. hand deliver comments to 11545 Rockville Pike, Rockville, Maryland, between 7:15 a.m. and 4:30 p.m. on Federal workdays. Copies of the comments received may be examined or copied for a fee at the NRC Public Document Room, located at 2120 L Street, NW (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Mr. David L. Gamberoni, M/S OWFN 12–E–4, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington DC, 20555. Telephone (301) 415–1144.

SUPPLEMENTARY INFORMATION: The Integrated Performance Assessment Process (IPAP) is a four phase process for systematically evaluating nuclear power plant licensee's safety performance. The IPAP also develops inspection recommendations that customize the inspection program for the next inspection period based on licensee strengths and weaknesses, and provides feedback to improve the effectiveness and implementation of regulatory programs.

The four IPAP phases include:

• Integrated Review of Licensee Performance

• Site Assessment Visit

• Final Analysis and Inspection Recommendation Development

• Assessment of Regulatory Programs

The NRC staff intends to make a brief presentation on the contents of the IPAP at the meeting. However, the main focus of the meeting will be to address any questions regarding the process and solicit comments. The NRC staff will consider the comments received during this public meeting as well as written comments in finalizing its recommendations to the Commission on the Integrated Performance Assessment Process.

Dated at Rockville, Maryland, this 2nd day of March 1995.

For the Nuclear Regulatory Commission.

Michael R. Johnson,

Chief, Performance Evaluation and Assessment Section, Inspection Program Branch, Directorate for Inspection and Support Programs, Office of Nuclear Reactor Regulation.

(Attachment to Notice of Meeting (IPAP))

Registration Form—United States Nuclear Regulatory Commission Integrated Performance Assessment Process Public Information Meeting TWFN, Auditorium, Rockville, Maryland

April 11, 1995.

Name ______________________

Title ______________________

Company/Organization ______________________

____

____

Address ______________________

____

____

Telephone Number ______________________

Comments ______________________

____

____

____

____

____

____

____

____

____

[FR Doc. 95–5775 Filed 3–8–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–245]

Northwest Utilities, Millstone Nuclear Power Station, Unit 1, License No. DPR–21; Receipt of Petition for Director's Decision Under 10 CFR 2.206

Notice is hereby given that by Petition dated January 15, 1995, Anthony J. Ross (Petitioner) has requested that the Nuclear Regulatory Commission take action with regard to Northeast Utilities. The Petition was submitted as a supplement to a letter submitted by the Petitioner on October 28, 1994, in which he requested that ''accelerated'' enforcement action be taken against Northeast Utilities for violations at Millstone involving procedure compliance, work control, and tagging control. As a basis for his request, the Petitioner alleges that since August 1993 violations in these areas have increased significantly when compared to previous like periods, that many of these violations have never been assigned a severity level, and that when the repetitive nature and duration of these violations are considered, and these violations are considered collectively together with violations that have been assigned a severity level, escalated enforcement action is warranted. By letter dated February 8, 1995, the Petitioner provided additional information in support of his Petition.

The request, as supplemented, is being treated pursuant to 10 CFR 2.206 of the Commission's regulations. The request has been referred to the Director of the Office of Nuclear Reactor Regulation.

A copy of the Petition and the February 8, 1995, supplement are available for inspection at the Commission's Public Document Room at 2120 L Street, NW., Washington, DC, and at the local public document room for Millstone Unit 1 located at the Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Dated at Rockville, Maryland, this 23rd day of February 1995.

For the Nuclear Regulatory Commission.

William T. Russell,

Director, Office of the Nuclear Reactor Regulation.

[FR Doc. 95–5772 Filed 3–8–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket No. 50–255]

Exemption

In the Matter of Consumers Power Co. (Palisades Plant).

I

The Consumers Power Company (the licensee) is the holder of Facility Operating License No. DPR–20, which authorizes operation of the Palisades Plant at a steady-state reactor power level not in excess of 2530 megawatts thermal. This facility consists of one pressurized water reactor located at the licensee's site in Van Buren County, Michigan. The license provides, among other things, that the licensee is subject to all rules, regulations, and orders of the Commission now or hereafter in effect.

II

The regulation 10 CFR 50.60, ''Acceptance Criteria for Fracture Prevention Measures for Light-water Nuclear Power Reactors for Normal Operation,'' 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 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. 10 CFR 50.60(b) specifies 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 low temperature overpressure transients that would produce pressure excursions exceeding the ASME Appendix G P/T limits while the reactor is operating at low temperatures, the licensee installed a low temperature overpressure (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 Appendix G P/T limits. To prevent the PORVs 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 maintain seal integrity of the 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 LTOP analysis indicates that using the ASME 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 rector coolant pumps during normal operation.

The licensee proposed in a letter dated February 10, 1995, that in determining the design setpoint for LTOP events for the Palisades Plant, the allowable pressure be determined using the safety margins developed in an alternate methodology in lieu of the safety margins currently required by Appendix G to 10 CFR part 50. Designated Code Case N-514, the proposed alternate methodology, is consistent with guidelines developed by the American Society of Mechanical Engineers (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. Code Case N–514, ''Low Temperature Overpressure Protection,'' has been approved by the ASME Code Committee but not yet approved for use in Regulatory Guide 1.147. 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 is revising 10 CFR 50.55a, which will endorse the 1993 Addenda and Appendix G of Section XI into the regulations.

An exemption from 10 CFR 50.60 is required to use the alternate methodology for calculating the maximum allowable pressure for the LTOP setpoint. By application dated February 10, 1995, the licensee requested an exemption from 10 CFR 50.60 for this purpose.

III

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person 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 or 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, 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 this appendix 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 the ASME Code.

Appendix G of the ASME Code requires that the P/T limits be calculated: (a) Using a safety factor of 2 on the principal membrane (pressure) stresses, (b) assuming a flaw at the surface with a depth of one-quarter of the vessel wall thickness and a length of 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 Palisades reactor vessel material.

In determining the setpoint for LTOP events, the licensee proposed to use safety margins based on an alternate methodology consistent with the proposed ASME Code Case 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% 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. Using the licensee's proposed safety factors instead of ASME Appendix G safety factors to calculate the LTOP setpoint will permit a higher LTOP setpoint than would otherwise be required, but will provide added margin to prevent normal operating surges from lifting the PORVs or cavitation of the reactor coolant pumps. Although this methodology would reduce the safety factor on the principal membrane stresses, the proposed criteria will provide adequate margins of safety to the reactor vessel during LTOP transients, thus providing an acceptable level of quality and safety. Accordingly, the use of the Code case will satisfy the underlying purpose of 10 CFR 50.60 for fracture toughness requirements for normal operation and anticipated operational occurrences.

IV

For the foregoing reason, the NRC staff has concluded that the licensee'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)(ii), such that application of 10 CFR 50.60 is not necessary in order to achieve the underlying purpose of this regulation.

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 the Consumers Power Company an exemption from the requirements of 10 CFR 50.60 such that in determining the setpoint for LTOP events, the ASME Appendix G curves for P/T limits are not exceeded by more than 10% in order to be in compliance with these regulations. This exemption is applicable only to LTOP conditions during normal operation.

Pursuant to 10 CFR 51.32, the Commission has prepared an environmental assessment and determined that the granting of this exemption will not have a significant effect on the quality of the human environment (February 27, 1995, 60 FR 10615).

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 2nd day of March 1995.

For the Nuclear Regulatory Commission.

Elinor G. Adensam,

Acting Director, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 95–5774 Filed 3–8–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 50–282 and 50–306]

Exemption

In the Matter of Northern States Power Co. (Prairie Island Units 1 and 2)

I

Northern States Power Company (NSP, the licensee) is the holder of Facility Operating Licenses Nos. DPR–42 and DPR–60 which authorize operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The units are pressurized water reactors (PWR) located in Goodhue County, Minnesota. The licenses provide, among other things, that the facilities are subject to all rules, regulations, and orders of the Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

II

Pursuant to 10 CFR 50.12(a), the NRC may grant exemptions from the requirements of the regulations (1) which are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) where special circumstances are present.

Section III.G.1 of Appendix R to 10 CFR part 50 requires, in part, that fire protection features shall be provided for structures, systems, and components important to safe shutdown so that one train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage. The staff has interpreted these provisions as requiring that features shall be such that one train of safe shutdown systems remains operable, notwithstanding a fire or consequences therefrom, without one having to perform any repair. In this context, the staff considers manually pulling fuses to isolate certain systems as a repair. Accordingly, the staff interprets Section III.G.1 of Appendix R as not permitting the pulling of fuses in order to be in compliance.

By letter dated May 2, 1994, the licensee requested an exemption to permit it to manually remove fuses from the power-operated relief valve control circuit in the event of a fire, in lieu of modifying plant hardware which would otherwise be required to achieve compliance with Section III.G.1 of Appendix R. The licensee's submittal initially referenced Section III.G.2 of Appendix R as providing the requirements from which the licensee was seeking an exemption, but in a follow-up telephone conversation with the staff the licensee concurred that Section III.G.1 is the appropriate reference.

This exemption was requested by the licensee in response to inspection findings identified in inspection reports 50–282/87–004, 50–282/88–013, 50–282/92–011 and 50–282/94–004. These findings addressed a concern with circuit failure modes that could adversely affect the ability to maintain hot shutdown in the event of a control room fire. This condition could occur if the power operated relief valves (PORV) block valves were not shut and a hot short damaged the PORV control circuit causing the PORV to open and remain open. Specifically, this involves the high/low pressure interface spurious signal concerns associated with Unit 1 PORVs CV–31231 and CV–31232 and their associated block valves MOV–32195 and MOV–32196 and with Unit 2 PORVs CV–31233 and CV–31234 and their associated block valves MOV–32197 and MOV–32198. As a precaution to prevent the potential loss of reactor coolant system (RCS) inventory during a control room fire, the licensee has proposed to close the PORV block valves prior to control room evacuation. The licensee also proposed to remove the PORV control circuit fuses to prevent a hot short or short to ground which may cause the PORV to open or be maintained open. As stated above, removal of fuses for isolation in such circumstances is considered a repair and, therefore, does not meet Appendix R, Section III.G.1, as interpreted by the staff.

The licensee's proposed actions of closing the PORV block valves and removing the control circuit fuses was reviewed by the staff and was found to be an effective means of assuring that a control room fire will not result in a sustained loss of RCS inventory.

The substance of the licensee's submittal was reviewed by Region III inspectors during the inspection conducted from July 18–22, 1988. The inspection findings were documented in NRC Inspection Report No. 50–282/88–013 and 50–306/88–013. The inspectors walked down the control room evacuation shutdown procedures. Step 3.3.1 of Procedure F5, Appendix B, ''Control Room Evacuation (Fire),'' directs the operators to remove/pull the fuses for the PORVs as an immediate action in response to a control room evacuation. The inspectors found that the fuse panels were readily accessible and the fuses were clearly identified in the panels. The inspectors also found that sufficient space is available to permit access for pulling fuses and that emergency lights and the fuse pullers had been provided in the vicinity of each panel. A training program has been established for all plant operators to enhance the familiarity with and proper response to the control room evacuation. Additionally, as a part of Emergency Operating Procedures (EOP) training, all the operators are trained on