to Facility Operating License No. DPR–50 for the Three Mile Island Nuclear Station, Unit No. 1, located in Dauphin County, Pa.

The proposed amendment requested approval of a revised reactor coolant maximum allowable dose equivalent iodine 131 specific activity level of 1.0 microcuries/gram.

The Commission had previously issued a Notice of Consideration of Issuance of Amendment published in the **Federal Register** on November 18, 1998 (63 FR 64118). However, by letter dated December 29, 1999, the licensee withdrew the proposed change request.

For further details with respect to this action, see the application for amendment dated October 19, 1998, as supplemented February 16, and September 2, 1999, and the licensee's letter dated December 29, 1999, which withdrew the application for license amendment. The above documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 7th day of February 2000.

For the Nuclear Regulatory Commission.

Timothy G. Colburn,

Sr. Project Manager, Section 1, Project Directorate I, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–3190 Filed 2–10–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–254 and 50–265]

Commonwealth Edison Company (Quad Cities Nuclear Power Station,Units 1 and 2);

Exemption

I.

The Commonwealth Edison Company (ComEd, the licensee) is the holder of Facility Operating Licenses Nos. DPR–29 and DPR–30 which authorize operation of the Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities). The license provides, among other things, that the facility 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 boiling water reactors (Units 1 and 2) located on the licensee's Quad Cities site in Rock Island County, Illinois. This exemption refers to both units.

II.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, requires that pressure-temperature (P–T) limits be established for reactor pressure vessels (RPVs) during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G states, ''The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.'' Appendix G of 10 CFRPart 50 specifies that the requirements for these limits are the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Appendix G Limits.

To address provisions of the proposed amendments to the technical specification (TS) P–T limits, the licensee requested in its submittal of November 12, 1999, that the staff exempt Quad Cities from application of specific requirements of 10 CFR Part 50, Section 50.60(a) and Appendix G, and substitute use of ASME Code Cases N–588 and N–640. Code Case N–588 permits the postulation of a circumferentially-oriented flaw (in lieu of an axially-oriented flaw) for the evaluation of the circumferential welds in RPV P–T limit curves. Code Case N–640 permits the use of an alternate reference fracture toughness (K_{IC} fracture toughness curve instead of K_{Ia} fracture toughness curve) for reactor vessel materials in determining the P–T limits. Since the pressure stresses on a circumferentially-oriented flaw are lower than the pressure stresses on an axially-oriented flaw by a factor of 2, using Code Case N–588 for establishing the P–T limits would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G and, therefore, an exemption to apply the Code Case would be required by 10 CFR 50.60. Likewise, since the K_{IC} fracture toughness curve shown in ASME Section XI, Appendix A, Figure A–2200–1 (the K_{IC} fracture toughness curve) provides greater allowable fracture toughness than the corresponding K_{Ia} fracture toughness curve of ASME Section XI, Appendix G, Figure G–2210–1 (the K_{Ia} fracture toughness curve), using Code Case N–640 for establishing the P–T limits would be less conservative than the methodology currently endorsed by 10 CFR Part 50, Appendix G and, therefore, an exemption to apply the Code Case would also be required by 10 CFR 50.60. It should be noted that, although Code Case N–640 was incorporated into the ASME Code recently, an exemption is still needed because the proposed P–T limits (excluding Code Cases N–588 and N–640) are based on the 1989 edition of the ASME Code.

Code Case N–588

The licensee has proposed an exemption to allow the use of ASME Code Case N–588 in conjunction with ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to determine the P–T limits.

The proposed amendments to revise the P–T limits for Quad Cities rely, in part, on the requested exemption. These proposed P–T limits have been developed using the postulation of a circumferentially-oriented reference flaw as the limiting flaw in a RPV circumferential weld in lieu of an axially-oriented flaw required by the 1989 Edition of ASME Section XI, Appendix G.

Postulating the Appendix G [axially-oriented flaw] reference flaw in a circumferential weld is physically unrealistic and overly conservative, because the length of the flaw is 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. Industry experience with the repair of weld indications found during preservice inspection, and data taken from destructive examination of actual vessel welds, confirms that any remaining flaws are small, laminar in nature, and do not transverse the weld bead orientation. Therefore, any potential defects introduced during the fabrication process, and not detected during subsequent nondestructive examinations, would only be expected to be oriented in the direction of weld fabrication. For circumferential welds this indicates a postulated defect with a circumferential orientation.

An analysis provided to the ASME Code's Working Group on Operating Plant Criteria (WGOPC) (in which Code Case N–588 was developed) indicated that if an axial flaw is postulated on a circumferential weld, then based on the stress magnification factors (M_m) given in the Code Case for the inside diameter circumferential (0.443) and axial (0.926) flaw orientations, it is equivalent to applying a safety factor of 4.18 on the pressure loading under normal operating conditions. Appendix G requires a safety factor of 2 on the contribution of the pressure load in the case of an axially-oriented flaw in an axial weld, shell plate, or forging. By postulating a circumferentially-oriented flaw on a circumferential weld and using the appropriate stress magnification factor, the margin of 2 is maintained for the contribution of the pressure load to the integrity calculation of the circumferential weld. Consequently, the staff determined that the postulation of an axially-oriented flaw on a circumferential RPV weld is a level of conservatism that is not required to establish P–T limits to protect the RCS pressure boundary from failure during hydrostatic testing, heatup, and cooldown.

The staff noted that ASME Code Case N–588 also includes changes to the methodology for determining the thermal stress intensity, K_{IT}, which was incorporated into Section XI of the ASME Code after the 1989 Edition. However, the licensee still used the methodology in the 1989 edition of the ASME Code to calculate K_{IT}. The staff already accepted the use of Code Case N–588 including the modifications made to the K_{IT} methodology for exemption requests by other licensees. Hence, the licensee may use the methodology in the 1989 Edition of ASME Section XI or the methodology contained in Code Case N–588 for determining K_{IT}.

In summary, the ASME Section XI, Appendix G, procedure was developed for axially-oriented flaws, which is physically unrealistic and overly conservative for postulating flaws of this orientation to exist in circumferential welds. Hence, the NRC staff concurs that relaxation of the ASME Section XI, Appendix G, requirements by application of ASME Code Case N–588 is acceptable and would maintain, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

Code Case N–640 (Formerly Code Case N–626)

The licensee has proposed an exemption to allow use of ASME Code Case N–640 in conjunction with ASME Section XI, 10 CFR 50.60(a) and 10 CFR Part 50, Appendix G, to determine P–T limits.

The proposed amendments to revise the P–T limits for Quad Cities rely in part on the requested exemption. These revised P–T limits have been developed using the K_{Ic} fracture toughness curve, in lieu of the K_{Ia} fracture toughness curve, as the lower bound for fracture toughness.

Use of the K_{Ic} curve in determining the lower bound fracture toughness in the development of P–T operating limits curve is more technically correct than use of the K_{Ia} curve since the rate of loading during a heatup or cooldown is slow and is more representative of a static condition than a dynamic condition. The K_{Ic} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a reactor vessel. The staff has required use of the initial conservatism of the K_{Ia} curve since 1974 when the curve was codified. This initial conservatism was necessary due to the limited knowledge of RPV materials. Since 1974, additional knowledge has been gained about RPV materials, which demonstrates that the lower bound on fracture toughness provided by the K_{Ia} curve is well beyond the margin of safety required to protect the public health and safety from potential RPV failure. In addition, P–T curves based on the K_{Ic} curve will enhance overall plant safety by opening the P–T operating window with the greatest safety benefit in the region of low temperature operations.

Since the RCS P–T operating window is defined by the P–T operating and test limit curves developed in accordance with ASME Section XI, Appendix G, continued operation of Quad Cities with these P–T curves without the relief provided by ASME Code Case N–640 would unnecessarily require the RPV to maintain a temperature exceeding 212 degrees Fahrenheit in a limited operating window during the pressure test. Consequently, steam vapor hazards would continue to be one of the safety concerns for personnel conducting inspections in primary containment. Implementation of the proposed P–T curves, as allowed by ASME Code Case N–640, does not significantly reduce the margin of safety and would eliminate steam vapor hazards by allowing inspections in primary containment to be conducted at lower coolant temperature. Thus, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the regulation will continue to be served.

In summary, the ASME Section XI, Appendix G, procedure was conservatively developed based on the level of knowledge existing in 1974 concerning RPV materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. The NRC staff concurs that this increased knowledge permits relaxation of the ASME Section XI, Appendix G, requirements by application of ASME Code Case N–640, while maintaining, pursuant to 10 CFR 50.12(a)(2)(ii), the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

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. The staff accepts the licensee's determination that the exemption would be required to approve the use of Code Cases N–588 and N–640. The staff examined the licensee's rationale to support the exemption requests and concurred that the use of the code cases would meet the underlying intent of these regulations. Based upon a consideration of the conservatism that is explicitly incorporated into the methodologies of 10 CFR part 50, appendix G; appendix G of the Code; and Regulatory Guide 1.99, Revision 2, the staff concludes that application of the code cases as described would provide an adequate margin of safety against brittle failure of the RPV. This is also consistent with the determination that the staff has reached for other licensees under similar conditions based on the same considerations. Therefore, the staff concludes that requesting exemption under the special circumstances of 10 CFR 50.12(a)(2)(ii) is appropriate and that the methodology of Code Cases N–588 and N–640 may be used to revise the P–T limits for Quad Cities Nuclear Power Station, Units 1 and 2.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the 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 Commonwealth Edison Company exemption from the requirements of 10 CFR Part 50, Section 50.60(a) and 10 CFR Part 50, Appendix G, for Quad Cities Nuclear Power Station, Units 1 and 2.

Pursuant to 10 CFR 51.32, an environmental assessment and finding of no significant impact has been prepared and published in the **Federal Register** (65 FR 5702). Accordingly, based upon the environmental assessment, the Commission has determined that the granting of this exemption will not result in any significant effect on the quality of the human environment.

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 4th day of February 2000.

For The Nuclear Regulatory Commission.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 00–3187 Filed 2–10–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–313]

Entergy Operations, Inc., Arkansas Nuclear One, Unit 1—Notice of Receipt of Application for Renewal of Facility Operating License No. DPR–51 for an Additional Twenty Year Period

The U.S. Nuclear Regulatory Commission has received an application from Entergy Operations, Inc., dated January 31, 2000, filed pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, and 10 CFR Part 54 for renewal of Facility Operating License No. DPR–51, which authorizes the applicant to operate Arkansas Nuclear One, Unit 1 (ANO–1). The current operating license for ANO–1 expires on May 20, 2014. ANO–1 is a pressurized-water reactor designed by Babcock and Wilcox and is located in Pope County, Arkansas. The acceptability of the tendered application for docketing and other matters, including an opportunity to request a hearing, will be the subject of a subsequent **Federal Register** notice.

A copy of the application is available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C. 20037.

Dated at Rockville, Maryland, this the fourth day of February 2000.

For the Nuclear Regulatory Commission.

Christoper I. Grimes,

Chief, License Renewal and Standarization Branch, Division of Regulatory Improvement Programs.

[FR Doc. 00–3186 Filed 2–10–00; 8:45 am]

BILLING CODE 7590–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 40–8968–ML]

In the Matter of: Hydro Resources, Inc. P.O. Box 15910, Rio Rancho, NM 87174; Notice of Appointment of Adjudicatory Employees

COMMISSIONERS:

Richard A. Meserve, Chairman
Greta J. Dicus
Nils J. Diaz
Edward McGaffigan, Jr.
Jeffrey S. Merrifield

Pursuant to 10 CFR 2.4, notice is hereby given that Messrs. William Von Till and John Lusher, Commission employees of the Office of Nuclear Material Safety and Safeguards, have been appointed as Commission adjudicatory employees within the meaning of section 2.4. Mr. Von Till will advise the Commission regarding issues related to the pending petition for review of LBP–99–30. Mr. Lusher will advise the Commission regarding issues related to the pending petition for review of LBP–99–19. Until such time as a final decision is issued in this matter, interested persons outside the agency and agency employees performing investigative or litigating functions in this proceeding are required to observe the restrictions of 10 CFR 2.780 and 2.781 in their communications with Messrs. Von Till and Lusher.

It is so ordered.

Dated at Rockville, Maryland, this 7th day of February, 2000.

For the Commission.

Annette Vietti-Cook,

Secretary of the Commission.

[FR Doc. 00–3191 Filed 2–10–00; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–344]

Portland General Electric Company (Trojan Nuclear Plant); Exemption

I.

Portland General Electric Company (licensee) is the holder of Facility Operating License No. NPF–1, which authorizes the licensee to possess the Trojan Nuclear Plant (TNP). The license states, in part, that the facility is subject to all the rules, regulations, and orders of the U.S. Nuclear Regulatory Commission (the Commission or NRC) now or hereafter in effect. The facility consists of a pressurized water reactor located at the licensee's site in Columbia County, Oregon. The facility is permanently shut down and defueled and the licensee is no longer authorized to operate or place fuel in the reactor.

II.

Section 50.54(p) of Title 10 of the Code of Federal Regulations states that ''The licensee shall prepare and maintain safeguards contingency plan procedures in accordance with appendix C of part 73 of this chapter for effecting the actions and decisions contained in the Responsibility Matrix of the safeguards contingency plan.''

Part 73 of Title 10 of the Code of Federal Regulations, ''PHYSICAL PROTECTION OF PLANT AND MATERIALS,'' states that ''This part prescribes requirements for the establishment and maintenance of a physical protection system which will have capabilities for the protection of special nuclear material at fixed sites and in transit and of plants in which special nuclear material is used.'' Section 73.55 of Title 10 of the Code of Federal Regulations, ''Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage,'' states that ''The licensee shall establish and maintain an onsite physical protection system and security organization which will have as its objective to provide high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.''

By letter dated January 27, 1993, the licensee informed the NRC that they no longer intend to operate the Trojan facility and intend to remove all spent nuclear fuel from the 10 CFR part 50 licensed site. By letter dated January 29, 1998, the licensee requested an exemption from the security requirements of 10 CFR 50.54(p) and 10 CFR part 73. 10 CFR 50.54(p) and 10 CFR 73.55 provide security requirements to protect the spent fuel while within the boundary of a licensed power reactor site. The requested exemption from the security requirements for the Trojan Nuclear Plant would be effective after the spent fuel has been removed from the reactor site by the licensee and relocated to the new independent spent fuel storage installation (ISFSI), which is not physically associated with the reactor site. The new ISFSI has been licensed under 10 CFR Part 72 for storage facilities not associated with a reactor site and possesses an approved physical plan as required by 10 CFR 72.180 and 10 CFR 73.51.