geometry system and to allow employees and contractors with unescorted access to keep their picture badges in their possession when leaving the Braidwood site.

IV

For the foregoing reasons, the NRC staff has determined that the proposed alternative measures for protection against radiological sabotage meet the same high assurance objective and the general performance requirements of 10 CFR 73.55. In addition, the staff has determined that the overall level of the proposed systems's performance will provide protection against radiological sabotage equivalent to that which is provided by the current system in accordance with 10 CFR 73.55.

Accordingly, the Commission has determined that, pursuant to 10 CFR 73.5, this 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 following exemption:

The requirement of 10 CFR 73.55(d)(5) that individuals who have been granted unescorted access and are not employed by the licensee are to return their picture badges upon exit from the protected area is no longer necessary. Thus, these individuals may keep their picture badges in their possession upon leaving the Braidwood site.

Pursuant to 10 CFR 51.32, the Commission has determined that the granting of this exemption will not result in any significant adverse environmental impact (60 FR 38855).

Dated at Rockville, Maryland, this 28th day of July 1995.

For the Nuclear Regulatory Commission.

Jack W. Roe,

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

[FR Doc. 95–19198 Filed 8–3–95; 8:45 am]

BILLING CODE 7590–01–P

[Docket Nos. 50–334 and 50–412]

Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, The Cleveland Electric Illuminating Company, The Toledo Edison Company (Beaver Valley Power Station, Units 1 and 2); Exemption

I

Duquesne Light Company, et al. (the licensee), is the holder of Operating License Nos. DPR–66 and NPF–73, which authorize operation of the Beaver Valley Power Station, Units Nos. 1 and 2, at steady state reactor core power levels not in excess of 2652 megawatts thermal (per unit). The licenses provide, among other things, that the licensee is subject to all rules, regulations and orders of the U.S. Nuclear Regulatory Commission (the Commission) now or hereafter in effect.

The facilities are two pressurized water reactors located at the licensee's site in Beaver County, Pennsylvania

II

Section 50.54(o) of 10 CFR part 50 requires that primary reactor containments for water cooled power reactors be subject to the requirements of appendix J to 10 CFR part 50. Appendix J contains the leakage test requirements, schedules, and acceptance criteria for tests of the leak tight integrity for the primary reactor containment and systems and components which penetrate the containment.

Section III.D.2(b)(ii) of appendix J to 10 CFR part 50 requires that an overall air lock Type B test shall be performed on air locks opened during periods when containment integrity is not required by the plant's Technical Specifications at the end of such periods at not less than P_a (the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases). The overall air lock Type B tests are intended to detect local leaks and measure leakage across each pressure-containing or leakage-limiting boundary of the air locks.

III

By letter dated February 4, 1994, the licensee requested an exemption to the requirements of Section III.D.2(b)(ii) of 10 CFR part 50, appendix J. The proposed exemption would permit local leak rate testing to be substituted for an overall air lock leakage test where the design permits. The exemption would be applicable to only those air lock components which are designed to be local leakage rate tested at a pressure of at least P_a. The leakage rate of each component would then be measured and verified to be within acceptable limits (i.e., containment leakage would be limited such that offsite radiation exposures will not exceed the guidelines of 10 CFR part 100 in the event of a design basis accident).

IV

The licensee presented information in support of its request for an exemption from the requirements of section III.D.2(b)(ii) of appendix J to 10 CFR part 50. The proposed exemption would allow maintenance to be performed on the air lock that could affect its sealing capability without requiring performance of the overall air lock leakage test. The licensee indicated that performance of the overall air lock test is very time consuming and results in additional occupational radiation exposure. The proposed exemption would allow local leakage testing to be substituted for the overall air lock leakage test when the design of the components permits local leakage rate testing at a pressure of at least P_a. A leakage rate would then be measured in accordance with the requirements of appendix J. The typical air lock components which could be tested in this manner are components such as the o-ring seals on the personnel air lock door(s), the mechanical penetrations for the 18-inch escape hatches, and the equalizing valves located on each of the air lock doors. 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 the airlock Type B testing is to ensure that each containment air lock will perform its safety function as part of the containment to control offsite radiation exposure resulting from a design basis accident. The proposed local leakage testing is sufficient to achieve the underlying purpose of the requirements of 10 CFR part 50, appendix J, section III.D.2(b)(ii) because it provides adequate assurance of the continued leak-tight integrity of the air lock(s). As a result, the application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

With respect to the requirements of 10 CFR 50.12(a)(1), the NRC staff has concluded that the requested action is authorized by law in that no prohibition of law exists which would preclude the activities which would be authorized by the exemption. In addition, for the reasons discussed above, the NRC staff has determined that the requested exemption does not present an undue risk to the public health and safety, is consistent with the common defense and security and that there are special circumstances present, as specified in 10 CFR 50.12(a)(2)(ii).

V

Based on the above, the NRC staff finds the requested exemption, to allow local leak rate testing to be substituted for an overall air lock leakage test where the design permits, acceptable. Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12(a), the requested exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The Commission finds that the special circumstances as required by 10 CFR 50.12(a)(2)(ii) are present.

An exemption is hereby granted from the requirements of section III.D.2(b)(ii) of appendix J to 10 CFR part 50, which requires an overall leakage test of air locks opened during periods when containment integrity is not required by the plant's Technical Specifications at the end of such periods at a pressure of not less than P_a. Local leak rate testing shall be substituted for the overall leakage test whenever this exemption is utilized.

Pursuant to 10 CFR 51.32, the Commission has determined that granting of this exemption will have no significant impact on the quality of the human environment (60 FR 30611).

This exemption is effective upon issuance.

Dated at Rockville, Maryland this 26th day of July 1995.

For the Nuclear Regulatory Commission.

Steven A. Varga,

Director, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 95–19199 Filed 8–3–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 50–280 and 50–281]

Virginia Electric and Power Company; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License Nos. DPR–32 and DPR–37 issued to the Virginia Electric and Power Company (the licensee) for operation of the Surry Power Station, Units 1 and 2 located in Surry County, Virginia.

The proposed amendment would incorporate revised pressure/temperature (P/T) limits and an associated Low Temperature Overpressure System (LTOPS) setpoint that will be valid to the end-of-license (28.8 and 29.4 effective full power years for Units 1 and 2, respectively). The proposed change also incorporates analytical and operational features into the Surry design basis on the P/T operating margin. The request also updates the unirradiated reactor vessel material toughness data presented in the Technical Specifications to reflect the data previously provided to the NRC in the licensee's response to Generic Letter 92–01, Revision 1, "Reactor Vessel Structural Integrity."

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Specifically, operation of Surry Power Station in accordance with the Technical Specification changes will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The safety analysis demonstrates that the proposed reactor vessel protection philosophy, and the associated pressure/temperature limits, LTOPS setpoint, and component operability requirements, ensure that reactor vessel integrity will be maintained during normal operation and design basis accident conditions. Specifically, adherence to the heatup/cooldown rate-dependent pressure/temperature operating limits ensures that the assumed design basis flaw will not propagate during normal operation. Below the LTOPS enabling temperature, automatic actuation of the PORVs ensures that the assumed design basis flaw will not propagate under design basis low-temperature overpressurization accident conditions. Above the enabling temperature, two pressurizer safety valves are sufficient to relieve the overpressurization due to the inadvertent startup of two charging pumps at water solid conditions without propagation of the assumed design basis flaw.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed Technical Specifications modify pressure/temperature operating limits, LTOPS setpoint and enabling temperature, and component operability requirements. The revised pressure/temperature operating limits and LTOPS setpoint are only slightly different than those currently in the Technical Specifications. The LTOPS enabling temperature remains unchanged. No operating limits or setpoints are added or deleted by the proposed changes. Therefore, it may be concluded that the operating limits and setpoint changes do not create the possibility of a new or different kind of accident. With regard to component operability requirements, restrictions on the number of charging pumps which may be operable, the number of PORVs which must be operable, and the allowable temperature difference between the steam generator primary and secondary remain unchanged. Only the setpoint temperature at which these restrictions apply have been modified. The proposed changes are entirely consistent with the reactor vessel integrity protection philosophy which ensures that the design basis reactor vessel flaw will not propagate under normal operation or postulated accident conditions. Further, the proposed changes do not invalidate . . . any component design criteria or the assumptions of any UFSAR Chapter 14 accident analysis.

3. Involve a significant reduction in a margin of safety. As described above, the reactor vessel integrity protection philosophy ensures that the design basis assumed flaw will not propagate under normal operation or design basis accident conditions. Adherence to the Technical Specification pressure/temperature operating limits ensures that the margin to vessel fracture provided by the ASME Section XI methodology is maintained. With regard to LTOPS protection, the safety analysis demonstrates that the proposed LTOPS design ensures margins consistent with those provided by ASME Section XI Appendix G methods as amended by ASME Code Case N–514. Utilization of ASME Code Case N–514 technically results in a reduction in the margin of safety, since a less restrictive LTOPS analysis design limit (i.e., 110% of the isothermal limit curve) is employed. However, the proposed design has been demonstrated to provide an acceptable margin of safety. Both industry experience and engineering evaluation support the conclusion that LTOPS design basis events may be expected to occur at essentially isothermal conditions. An engineering evaluation demonstrates that any reduction in allowable pressure due to thermal stresses which may be expected to exist during an LTOPS design basis event is insignificant when compared to margins provided by the ASME Section XI Appendix G methods for calculating pressure/temperature operating limits. This design maximizes the operating margin above the minimum RCS pressure for