ID950005 (Feb. 10, 1995)
ID950013 (Jul. 28, 1995)
ID950014 (Jul. 28, 1995)
Oregon
OR950001 (Feb. 10, 1995)
OR950004 (Feb. 10, 1995)
OR950017 (Dec. 15, 1995)
Washington
WA950001 (Feb. 10, 1995)
WA950002 (Feb. 12, 1995)
WA950003 (Feb. 12, 1995)
WA950007 (Feb. 12, 1995)
WA950008 (Feb. 12, 1995)
Wyoming
WY950004 (Feb. 10, 1995)
WY950008 (Feb. 10, 1995)
WY950009 (Feb. 10, 1995)
WY950011 (Feb. 10, 1995)
WY950021 (Feb. 10, 1995)
WY950023 (Feb. 10, 1995)
WY950024 (Feb. 10, 1995)

General Wage Determination Publication

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Signed at Washington, D.C. this 26th day of January 1996.

Philip J. Gloss,

Chief, Branch of Construction Wage Determinations.

[FR Doc. 96–1913 Filed 2–1–96; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

[Docket No. 50–440]

The Cleveland Electric Illuminating Company, et al.; 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 No. NPF–58 issued to The Cleveland Electric Illuminating Company, et al. (the licensee), for operation of the Perry Nuclear Power Plant, Unit No. 1 located in Lake County, Ohio.

The proposed amendment would change the Technical Specification surveillance frequency for the drywell bypass leakage rate test from 18 months to 120 months (10 years) with a more frequent testing requirement if performance degrades. Additionally, specific leakage limits would be deleted for the air lock seal and barrel tests. Also, surveillance frequencies for the air lock interlock test and seal pneumatic system leak test would be changed from 18 months to 24 months. Finally, the surveillance frequencies for the air lock barrel test would be changed from ''each COLD SHUTDOWN if not performed within the previous 6 months'' to ''at least once per 24 months'' and from 18 months to 24 months. The licensee requested that this amendment be approved for use during the current refueling outage which began on January 27, 1996.

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:

I. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes in frequency for the drywell bypass leakage and drywell air lock surveillances will continue to ensure that no paths exist through drywell boundary components that would permit gross leakage from the drywell to bypass the containment pressure-suppression feature (the suppression pool) and result in exceeding the primary design basis limit. The Mark III primary containment system satisfies General Design Criterion 16 of Appendix A to 10 CFR Part 50. Maximum drywell bypass leakage was determined previously by reviewing the full range of postulated primary system break sizes. The limiting case was a primary system small break LOCA that yielded a design allowable drywell bypass leakage rate limit of approximately 58,000 scfm. The Technical Specification acceptable limit for the bypass leakage following a surveillance is less than 10% of the design basis value. The most recent bypass leakage value was approximately 0.2% of the design allowable leakage rate limit for the limiting event. Programmatic and oversight controls are maintained that ensure drywell bypass leakage remains a fraction of the design allowable leakage limit.

The drywell is exposed to essentially 0 psig during normal plant operation and 2.5 psig during drywell bypass leak rate testing. These pressures are considerably lower than the structural integrity test pressure and are not likely to initiate a crack or cause an existing crack to grow. Visual inspections of the accessible drywell surfaces that have been performed since the structural integrity tests have not revealed the presence of abnormal cracking or other abnormalities. Therefore, drywell degradation is not expected due to testing or operation and it is not considered credible for the passive drywell structure to begin to leak sufficiently to impact the design drywell bypass leakage limit.

The primary containment's ability to perform its safety function is fairly insensitive to the amount of drywell bypass leakage, thereby providing a margin to loss of the drywell safety function that is not normally available for safety systems. This insensitivity is demonstrated by the extremely high limiting event design basis allowable leakage for the drywell (approximately 58,000 scfm as discussed above). An even higher allowable leakage can be accommodated by the primary containment due to containment design margin. It would take valves in multiple penetration flow paths leaking excessively to cause the primary containment to fail as a result of overpressurization. Therefore, the probability that drywell isolation valve leakage will result in primary containment failure due to excessive drywell bypass leakage is not significant and this drywell/primary containment failure mode is not credible.

The proposed Technical Specification changes have no significant impact on the IPE conducted in accordance with NRC Generic Letter 88–20. The IPE considered primary containment overpressurization failure as part of the primary containment performance assessment. Due to the magnitude of acceptable drywell bypass leakage and the extremely low probabilities of experiencing excessive leakage, preexisting excessive drywell bypass leakage was considered a non-significant contributor to primary containment failure. In a beyond-design-basis ''severe accident,'' the surveillance frequencies for the air lock failure can occur with or without preexisting excessive drywell bypass leakage. This is due to physical phenomena associated with potentially extreme environmental conditions inside primary containment following a severe accident. However, the calculated frequency of such extreme conditions is very small. The proposed changes do not impact the IPE evaluated phenomena causing primary containment overpressurization failure and do not significantly increase the probability that the drywell has preexisting excessive leakage. The proposed changes therefore, would not contribute to these accident scenarios.

The movement of the air lock leakage rate tests to the Drywell Specification and the elimination of the Notes in the Improved Technical Specifications are proposed because drywell leakage rate requirements are the essence of drywell operability. Leakage rates discovered outside limits will always clearly result in entering the actions for drywell inoperability. Additionally, the requirements for the drywell air lock seal and barrel tests to meet specific leakage limits are deleted since the ability of the drywell to perform its safety function is not dependent on the air lock meeting a specific leakage limit. The limiting case for drywell bypass leakage is based on total leakage through all drywell paths other than the suppression pool vents. Total drywell bypass leakage from such paths (including the air lock) should not exceed the acceptable design limit of drywell bypass leakage. The proposed Technical Specifications will still require performance of seal and barrel leak tests. Additionally, the proposed changes include minor administrative changes which clarify the requirement format or change the requirement to match the plant design bases.

For the reasons discussed above, the proposed changes do not have any significant risk impact to accidents previously evaluated and do not significantly increase the consequences of an accident previously evaluated. Additionally, drywell bypass leakage is not the initiator of any accident evaluated; therefore, changes in the frequency of the surveillance for drywell bypass leakage does not increase the probability of any accident evaluated.

II. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will impact the test frequencies and will not result in any change in equipment response in the unlikely event of an accident. The changes do not alter equipment design or capabilities. The changes do not present any new or additional failure mechanisms. The drywell is passive in nature and the surveillance will continue to verify that its integrity has not degraded. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. The proposed change does not involve a significant reduction in a margin of safety.

Drywell integrity and reliability have been demonstrated during past drywell bypass leakage surveillances. Appropriate design basis assumptions will be maintained. Drywell integrity will continue to be tested by the proposed periodic drywell bypass leakage test, the drywell air lock door latching and interlock mechanism surveillance, and additional surveillances including exercising the drywell isolation valves. In combination, these surveillances will provide adequate assurance that drywell bypass leakage will not exceed the design basis limit. Margins of safety will not be reduced. Therefore, the proposed change does not cause a reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in preventing startup of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By March 4, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Perry Public Library, 3753 Main Street, Perry, Ohio. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1–(800) 248–5100 (in Missouri 1–(800) 342–6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to Gail H. Marcus: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Jay E. Silberg, Shaw, Pittman, Potts & Trowbridge, 2300 N Street NW., Washington, DC 20037, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated January 16, 1996, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Perry Public Library, 3753 Main Street, Perry, Ohio.

Dated at Rockville, Maryland, this 29th day of January 1996.

For the Nuclear Regulatory Commission.

Jon B. Hopkins, Sr.,

Project Manager, Project Directorate III–3, Division of Reactor Projects—III/IV, Office of Nuclear Reactor Regulation.

[FR Doc. 96–2206 Filed 2–1–96; 8:45 am]

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[Docket No. 50–331]

IES Utilities, Inc; 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 No. DPR–49 issued to IES Utilities Inc. for operation of the Duane Arnold Energy Center (DAEC) located in Palo, Iowa.

The proposed amendment would modify the requirements for testing an emergency diesel generator (EDG) when the other is inoperable. The amendment would correct an editorial error in the DAEC Operating License and would correct an erroneous reference in the Technical Specifications (TS).

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:

1. The proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes are administrative in nature and are consistent with previously-published NRC guidance. The proposed revision does not change any accident analysis, plant safety analysis or calculations; degrade existing plant programs; or modify any functions of safety related systems or accident mitigation functions for which the DAEC has previously been credited. The proposed revision to the Surveillance Requirements will continue to assure OPERABILITY as required, but eliminate unnecessary operation of an EDG and is consistent with the requirements of the Improved Standard TS, NUREG–1433.

2. The proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revision does not