June 8, 1999

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Mr. Eliot Protsch
President
IES Utilities Inc.
200 First Street, SE.
P.O. Box 351
Cedar Rapids, IA 52406-0351

SUBJECT: DUANE ARNOLD ENERGY CENTER - ISSUANCE OF AMENDMENT RE: SPENT FUEL RACKS STORAGE UPDATE (TAC NO. MA4658)

Dear Mr. Protsch:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 226 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 22, 1999.

The amendment revises DAEC TS Section 4.3, "Fuel Storage," by updating the criticality requirements (k-infinity and U-235 enrichment limits) for storage of fuel assemblies in the spent fuel racks. This change allows for storage of nuclear fuel assemblies with new designs, including GE-12 with a 10X10 pin array.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
*Original signed by*
Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 226 to License No. DPR-49
2. Safety Evaluation



cc w/encls: See next page

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|---|---|---|---|---|---|
| NAME | BMozafari | EBarnhill | GHubbard | AHodgdon | CCraig |
| DATE | 6/3/99 | 6/3/99 | 05/18/99 | 05/24/99 | 6/3/99 |

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 8, 1999

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President
IES Utilities Inc.
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Brenda Mozafari

Brenda L. Mozafari, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-331

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 226
License No. DPR-49

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by IES Utilities Inc., et al., dated January 22, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.226, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Claudia M. Craig

Claudia M. Craig, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 8, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 226

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised areas are identified by amendment number and contain marginal lines indicating the areas of change.

| Remove | Insert |
|---|---|
| 4.0 - 2 | 4.0 - 2 |

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having the following limits for maximum k-infinity in the normal reactor core configuration at cold conditions and maximum lattice-average U-235 enrichment weight percent:

| | | k-∞ | wt % |
|---|---|---|---|
| i) | 7x7 and 8x8 pin arrays (Holtec and PaR racks) | ≤ 1.31 | ≤ 4.6 |
| ii) | 9x9 and 10x10 pin arrays (Holtec racks) | ≤ 1.29 | ≤ 4.95 |
| iii) | 9x9 and 10x10 pin arrays (PaR racks) | ≤ 1.39 | ≤ 4.95 |

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in 9.1 of the UFSAR; and

c. A nominal 6.060 inches for HOLTEC designed and 6.625 inches for PaR designed center to center distance between fuel assemblies placed in the storage racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum k-infinity of 1.31 in the normal reactor core configuration at cold conditions;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.90$ if dry, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 6.625 inch center to center distance between fuel assemblies placed in storage racks.

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 226 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.

CENTRAL IOWA POWER COOPERATIVE

CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

## 1.0 INTRODUCTION

By letter dated January 22, 1999, IES Utilities Inc. (licensee) requested changes to the Duane Arnold Energy Center (DAEC) Technical Specifications (TS) to update the criticality requirements (k-infinity and U-235 enrichment limits) for storage of fuel assemblies in the spent fuel racks. The change would allow for storage of fuel assemblies with new designs, including GE-12 with a 10x10 fuel rod array.

## 2.0 EVALUATION

The DAEC spent fuel pool includes storage racks designed by the PaR Corporation and by Holtec International. The PaR racks are composed of 0.125-inch thick aluminum boxes with a 0.080-inch thick boral absorber panel located between boxes in a 0.2185-inch cavity. The fuel assemblies have a lattice spacing of 6.625 inches and the boral absorbers have a minimum loading of 0.0232 g B-10/sq cm and are clad on both sides with 0.0175-inch thick aluminum. The Holtec racks consist of 0.060-inch thick stainless steel boxes on a 6.06-inch lattice spacing with a 5.90-inch inside opening. The 0.070-inch thick boral absorber has a nominal loading of 0.0162 g B-10/sq cm. The DAEC spent fuel storage racks are presently licensed for storage of boiling-water reactor (BWR) fuel assemblies having a maximum k-infinity (k-inf) of 1.31 in the normal reactor core configuration at cold conditions and an average U-235 enrichment of 4.6 weight percent (w/o).

The analysis of the reactivity effects of fuel storage in the DAEC racks was performed with both the CASMO3 two-dimensional transport theory code and the NITAWL-KENO5a three-dimensional Monte Carlo code package using the 238-group SCALE cross-section library. Independent check calculations were made with the MCNP Monte Carlo code. CASMO3 was used to determine the peak reactivity over burnup and to evaluate small reactivity increments associated with manufacturing tolerances and pool temperature changes. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results

from numerous critical experiments. These experiments simulate the DAEC spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, the two independent methods of analysis (MCNP and KENO5a) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the DAEC storage racks with a high degree of confidence.

The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) Racks contain most reactive fuel for the case being analyzed, without any control rods or any burnable poison, except gadolinia, as appropriate.

(2) Unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures.

(3) Assumption of infinite array (no neutron leakage) of storage cells in the radial direction.

(4) Neutron absorption in minor structural material is neglected (i.e., spacer grids are analytically replaced by water).

(5) The fuel assemblies were evaluated using a uniform average (planar) enrichment and do not include the natural $UO_2$ blankets at each end.

(6) The flow channel was homogenized with the immediately surrounding water in the CASMO3 model.

The staff concludes that appropriately conservative assumptions were made.

The following General Electric Company fuel assembly types were used for the criticality analyses:

(1) GE-10, 8x8 assembly with a single large water hole replacing 4 fuel rods

(2) GE-13, 9x9 assembly with 2 water holes replacing 7 fuel rods

(3) GE-12, 10x10 assembly with 2 large water holes replacing 8 fuel rods

The design basis reactivity calculations for the Holtec racks accounted for uncertainties and allowances previously reviewed and evaluated by the NRC for the Holtec racks during the spent fuel pool storage capacity expansion in 1994 which remain applicable. These values included manufacturing tolerances, flow channel bulging, and fuel enrichment and density. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations.

The design basis reactivity calculations for the PaR racks accounted for uncertainties due to manufacturing tolerances in boron loading, boral width, clad thickness, lattice spacing, fuel enrichment and density, water hole thickness, and eccentric assembly position, as well as a calculational bias and uncertainty which were determined from benchmark calculations. At the very low burnups required for the higher enrichment fuel, depletion calculational uncertainties as well as bulging of the flow channel were found to be insignificant.

The current fuel assemblies at DAEC have initial maximum lattice enrichments less than 3.7 w/o U-235. In BWR fuel, there is a need for distributed enrichments to avoid power peaking problems, and, since 5.0 w/o is the maximum enrichment allowed for any single fuel rod, it is not likely that a BWR assembly will exceed an average enrichment of about 4.6 w/o U-235. Therefore, calculations were made for the fuel designs at DAEC assuming an average enrichment of 4.6 w/o U-235 in both the spent fuel storage rack configurations and the DAEC core geometry (6.0-inch assembly pitch, 20°C). The results indicate that any of the fuel types with an average initial enrichment of 4.6 w/o or less and a k-inf in the standard core geometry less than or equal to 1.40 would result in a rack effective multiplication factor (k-eff) of less than 0.95, including all appropriate uncertainties at a 95% probability, 95% confidence (95/95) level. This meets the staff's criterion for spent fuel pool storage and is, therefore, acceptable. Therefore, the current TS requirement for a maximum k-inf of 1.31 and an average U-235 enrichment of 4.6 w/o remain valid.

Similar calculations for fuel initially enriched to 4.95 w/o U-235 in the PaR racks, which include the GE-12, 10x10 array and the GE-13, 9x9 array, show that any assembly with a DAEC core k-inf of 1.39 or less would meet the 0.95 k-eff acceptance criterion for storage in the PaR racks, regardless of burnup.

Calculations for fuel initially enriched to 4.95 w/o U-235 in the Holtec racks, which also include the GE-12, 10x10 array and the GE-13, 9x9 array, show that any DAEC assembly with a core k-inf of 1.29 or less would be acceptable for storage regardless of burnup.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events due to temperature and water density effects, abnormal or eccentric fuel assembly positioning, and the drop of a fuel assembly on top of the storage rack which could lead to an increase in reactivity. However, such events were found to have a negligible effect and the resulting reactivity would remain below the 0.95 design basis for both the PaR and the Holtec storage racks.

The following changes to TS 4.3.1 have been proposed as a result of the requested criticality requirement update and are acceptable.

Fuel assemblies having the following limits for maximum k-inf in the normal reactor core configuration at cold conditions and maximum lattice-average U-235 enrichment would be acceptable for storage in the DAEC spent fuel storage racks:

| | K-inf | weight percent U-235 |
|---|---|---|
| 1) 7x7 and 8x8 fuel rod arrays (Holtec and PaR racks) | ≤1.31 | ≤4.6 |
| 2) 9x9 and 10x10 fuel rod arrays (Holtec racks) | ≤1.29 | ≤4.95 |
| 3) 9x9 and 10x10 fuel rod arrays (PaR racks) | ≤1.39 | ≤4.95 |

Based on the review described above, the staff finds that the criticality aspects of the proposed update to the DAEC spent fuel storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (64 FR 9192). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Kopp, SRXB

Date: June 8, 1999