February 2, 1994

Docket No. 50-331

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Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

SUBJECT: AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-49 (TAC NO. M86284)

The Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications in response to your application dated March 26, 1993 and as supplemented on September 15 and November 23, 1993 and January 10, 1994 and changes license conditions to incorporate reference to the updated Final Safety Analysis Report and correct a typographical error.

The amendment revises the Technical Specifications to allow reracking the DAEC spent fuel pool with high density fuel storage racks.

A copy of the Safety Evaluation and the Notice of Issuance are also enclosed. The original of the notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed by

Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 195 to License No. DPR-49
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

LA:PD3-3/DRPW	PM:PD3-3:DRPW	D:PD3-3	D:SPLB	D:PRPB
MRushbrook	RPulsifer:sw	JHannon	CMcCracken	LCunningham
1/21/94	1/21/94	1/21/94	1/25/94	2/1/94
D:SRXB	D:ECGB	OGC-WF1		
RJones	GBagchi			
1/26/94	2/1/94	/ /94		

@ concurred 1/31/94 on another copy. Also see changes there. AP-H

DOCUMENT NAME: G:\DUANEARN\DUA86284.AMD

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AP-H 2/01/94

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

February 2, 1994

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc
Post Office Box 351
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Robert M. Pulsifer, Project Manager
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Office of Nuclear Reactor Regulation

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1. Amendment No. 195 to
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See next page

Duane Arnold Energy Center
Iowa Electric Light and Power Company

cc:

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

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

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light and Power Company, et al., dated March 26, 1993 and as supplemented on September 15 and November 23, 1993 and January 10, 1994 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, Facility Operating License No. DPR-49 is amended to reflect the updated Final Safety Analysis Report in condition 2.B.(2) and correct a typographical error in condition 2.B(4).

Revise paragraph 2.B.(2) to read:

2.B.(2) IELP, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended as of June 1992 and as supplemented by letter dated March 26, 1993.

Revise paragraph 2.B.(4) to read:

2.B.(4) to read:

IELP, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or association radioactive apparatus components.

The license is also 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. DRP-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 195, 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 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: February 2, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

LIST OF AFFECTED PAGES

REMOVE	INSERT
5.5-1	5.5-1
5.5-2	-----

5.5 SPENT AND NEW FUEL STORAGE

1. The new fuel storage facility shall be such that the effective neutron multiplication factor (k_{eff}) of the fuel, dry is less than 0.90 and flooded is less than 0.95. These k_{eff} values are satisfied if the maximum infinite lattice multiplication factor (k_∞) of the individual fuel bundles is ≤ 1.31.

2. The k_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95. This k_{eff} value is satisfied if the maximum, exposure-dependent k_∞ of the individual fuel bundles is ≤ 1.31 and the initial uniform average enrichment is ≤ 4.6 wt% U-235.

3. Spent fuel shall only be stored in the spent fuel pool in a vertical orientation in approved storage racks.

Bases

The basis for the k_∞ limit is described in Reference 1 for the GE-designed new fuel storage racks. Compliance with this specification is demonstrated by comparing the beginning-of-life, uncontrolled k_∞ values for the fuel type of interest to the 1.31 limit. For GE-supplied fuel, k_∞ values can be found in Reference 2. The k_∞ values found in Reference 2 represent the maximum, exposure-dependent lattice reactivity and can be conservatively applied to the new fuel limit.

Calculations have been performed (Reference 3) to determine the bounding reactivity limits for bundles of GE-designed fuel, when stored in the spent fuel storage racks of an approved design. These analyses were performed conservatively assuming uniform average initial enrichments in a parametric evaluation for fuel with enrichments up to 4.6 wt% U-235 initially. The bounding limit of an infinite multiplication factor of 1.31 for fuel of 4.6 wt% enrichment (or less) was evaluated at the maximum k_∞ over burnup and includes a conservative allowance for possible differences between the rack design calculations and the fuel vendor calculations.

References

1) General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A.*

2) General Electric Fuel Bundle Designs, NEDE-31152-P.*

3) Licensing Report for Spent Fuel Storage Capacity Expansion, Duane Arnold Energy Center, Holtec Report HI-92889.

*Latest NRC-approved revision.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 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 March 26, 1993, and supplemented on September 15, November 23, 1993, and January 10, 1994, IES Utilities Inc. (or the licensee) requested to amend the Operating License (OL) and Technical Specifications (TSs) for Duane Arnold Energy Center (DAEC). The proposed amendment is intended to permit expansion of the spent fuel pool (SFP) storage capacity from its current licensed capacity of 2050 assemblies to 3152 assemblies. The licensee intends to accomplish the proposed expansion by replacing the existing SFP racks with maximum density racks and by providing additional storage capacity of no more than 323 cells in a proposed fuel rack storage module to be located in the cask loading area of the cask pit. The cask pit rack will be used as a means to retain full-core offload capability after such capacity is exhausted in the SFP itself. The licensee intends to complete the rerack operation in three campaigns over a fifteen year period; the first campaign scheduled to begin in 1994 will increase the storage capacity to 2411 assemblies.

The licensee has determined that, in its current configuration, the SFP will reach its capacity by the year 2001, with the loss of the capacity to completely off-load the reactor core in 1998. The licensee has reviewed many options and has concluded that reracking the SFP was the best available option to increase the SFP storage capacity and allow the facility to continue to operate.

2.0 EVALUATION

2.1 HEAVY LOADS AND THERMAL-HYDRAULICS ASPECTS

This section of the safety evaluation addresses the thermal-hydraulic and heavy loads issues related to the reracking, assuming the cumulative storage capacity of the three reracking campaigns (3152 storage cells).

The licensee provided the staff a licensing report for the proposed SFP rerack. This report was prepared by Holtec International and was submitted as Attachment 3 to the letter from the licensee dated March 26, 1993. The licensee clarified some aspects of their submittal during a technical meeting with the staff and the licensee's contractor Holtec International on July 8, 1993.

2.1.1 Background of Control of Heavy Loads

DAEC SFP is licensed to contain 20 fuel storage racks, 19 of which are currently installed, for a total of 2050 storage cells with all storage racks installed. During the period in which the first rerack campaign will be conducted, approximately 1280 of the 2050 storage cells will be occupied with spent fuel. The licensee has determined that a sufficient number of cells exist to permit the relocation of all fuel such that the existing rack modules can be emptied and removed from the pool, and the new rack modules installed in a programmed manner without the need to move the rack modules over any region of the pool containing fuel. The new rack modules will not be anchored to the pool floor.

In the licensing report prepared for the licensee by Holtec International, the licensee committed to using a remotely engaging lifting device meeting the criteria of Section 5.1.6(1)(a) of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980, and that complies with all provisions of ANSI 14.6 - 1978, including the primary stress criteria. The licensee has also committed to developing safe load paths for the rerack, providing comprehensive training for the installation crew, and developing a complete set of procedures that comply with Section 5.1.1 of NUREG-0612 covering all aspects of the reracking process.

The Reactor Building Crane will be used in the reracking operation. A new single-failure-proof crane trolley and main hoist that meets the guidelines of Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," and NUREG-0554, "Single Failure Proof Cranes for Nuclear Power Plants," was installed in 1985. The existing bridge, trolley, and hoist controls as well as the bridge itself were not replaced. A seismic evaluation of the new trolley and existing bridge structure has been conducted by the licensee, but has not been submitted for review by the staff. The main hoist has a 100-ton load capacity and has undergone a 125-ton static load test. The maximum weight of any existing or replacement storage rack and its associated handling tool is 12 tons. The licensee has committed to conducting a preventive maintenance checkup and inspection of the reactor building crane prior to the reracking.

The licensee has developed a fuel movement strategy, discussed during their presentation to the staff on July 8, 1993, that places spent fuel cells in specific locations in the pool such that no heavy loads will be carried directly over irradiated fuel. During the modification phase of the reracking project, administrative controls governing safe load paths will supplant the Reactor Building Crane interlocks and limit switches. Similar controls were successfully employed during a previous reracking project at DAEC in 1979. In addition, the licensee committed to preventing the movement of any load weighing more than the combined weight of a fuel bundle and grapple in the spent fuel pool area until all the fuel in the pool has decayed for a minimum of three months. The licensee concludes that this provides sufficient time for decay of gaseous radio-nuclides in the fuel such that the assumed release of gases from damage to all stored fuel assemblies due to a potential heavy loads drop would result in a potential offsite dose less than 10% of the 10 CFR 100 limits.

In a letter dated September 15, 1993, the licensee committed to prevent the transfer of spent fuel that has decayed less than 5 years to the Cask Pit Rack. This commitment is in addition to the commitments made in the letter dated March 26, 1993, that include sealing the Cask Pit floor drain, preventing the installation of the gate between the Cask Pit and the Spent Fuel Pool, and preventing the transfer of heavy loads over the Cask Pit if it is utilized to store spent fuel.

2.1.2 Evaluation of Heavy Loads

The design of the Ederer hoist and trolley system to be used by the licensee was evaluated in a staff SER of the Generic Licensing Topical Report EDR-1, Rev. 3, for Ederer's Nuclear Safety-Related Extra Safety and Monitoring (X-SAM) cranes, dated August 3, 1983. The SER documented that the design of the main hoist and trolley complies with the criteria for single-failure proof cranes presented in NUREG-0554. However, since the licensee installed the trolley system on an existing bridge, the licensee was required to perform a seismic analysis to determine whether the bridge and trolley system meet the seismic analysis guidance of Regulatory Guide 1.29. While the licensee has performed an analysis concluding it met seismic requirements, it was not submitted to the staff for review. Therefore, the crane system cannot be considered as a single failure proof crane system.

During the reracking process, the licensee has committed to use a single-failure-proof lifting rig designed to meet the criteria of Section 5.1.6(1) of NUREG-0612. Lifting rigs that conform to the criteria of NUREG-0612 for single-failure-proof handling systems satisfy the regulatory guidance of Regulatory Guide 1.13 and Section 9.1.5 of the SRP, and the requirements of the General Design Criteria 4 and 61 of Appendix A to 10 CFR Part 50 with regard to the design of heavy load handling systems.

The licensee has committed to implement operator training programs, crane inspection and maintenance, safe load paths, and comprehensive procedures for the reracking operation which complies with the criteria of 5.1.1 of NUREG-0612.

In addition to imposing administrative restrictions on the handling of heavy loads near spent fuel, the licensee has also provided evidence of meeting the general guidelines of Section 5.1 of NUREG-0612. The licensee analyzed a postulated load drop event involving a spent fuel pool rack assembly and found by analysis that the load drop would not impair the structural integrity of the SFP. The licensee has committed not to carry heavy loads in the spent fuel pool area until all fuel in the pool has decayed such that a release from damage to all stored fuel assemblies would result in offsite doses less than 10% of the 10 CFR 100 limits. These analyses and administrative restrictions provide assurance that an uncontrolled decrease in pool cooling water inventory would not result from a postulated load drop, and that releases of radioactive material from postulated load drops are well within 10 CFR 100 limits.

While the licensee's crane system cannot be considered single failure proof, the licensee's use of a single failure proof hoist and trolley system and a single failure proof lifting rig; commitment to operator training programs, crane inspection and maintenance, safe load paths, and reracking procedures; and performance of a load drop analysis provides adequate assurance that the licensee's planned actions are consistent with the "defense-in-depth" approach to safety described on NUREG-0612 and the requirements of GDCs 4, 61, and 62. Therefore, the staff finds the licensee's program is acceptable.

2.1.3 Background for Thermal-Hydraulics

The Spent Fuel Pool Cooling and Cleanup System (FPCCS) consists of two parallel loops, each consisting of one full capacity pump, one filter-demineralizer and one $4x10^6$ Btu/hr heat exchanger. Cooling water is supplied to the heat exchangers from the Reactor Building Closed Loop Cooling Water System (RBCCW). The heat exchangers in the RHR system are used in conjunction with the FPCCS to supplement pool cooling when the RHR system is not needed for cooling the reactor cavity. Makeup water to the FPCCS is normally provided from the Condensate System, with alternate makeup capacity provided through a hose connection from the Emergency Service Water System (ESW). Makeup can also be supplied directly to the SFP through fire water hoses.

The Holtec Licensing Report stated that the decay heat load calculation for the SFP was performed in accordance with the provisions of Branch Technical Position ASB-9.2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, July 1981. The licensee considered four discharge scenarios to evaluate the total decay heat load. For all scenarios, fuel assemblies located in the SFP were assumed to have accumulated 4.5 years of full power operation.

Scenarios (1) and (2) were presented to demonstrate compliance with the provisions of Section 9.1.3 of NUREG-0800, "Standard Review Plan," but were modified to reflect the DAEC 18-month operating cycle. Scenarios (3) and (4) analyze discharge sequences that more closely correspond to the licensee's refueling and abnormal discharge practices. In scenario (3), the heat load calculation for a normal full core offload sequence with both cooling trains in operation bounds the SRP full core offload analysis. The calculations assumed a total of 3152 locations for fuel storage and was carried out at the point in time when the addition of a normal batch to the pool will leave it with insufficient capacity to accept another normal batch, while still maintaining full core offload capability. The discharge consisted of first offloading the entire core to the pool followed by the re-transfer of all assemblies back to the reactor vessel except the "burned" normal batch (116 assemblies). Fuel assembly transfer begins 120 hours after shutdown and is conducted at a rate of 144 assemblies per 24 hours. Scenario (4) assumes the refueling described in scenario (3) has occurred, and that following restart, a full core offload occurs after 36 days of operation. The duration of the refueling shutdown described in scenario (3) is assumed to be 45 days. The licensee found that the heat load calculated during scenario (4) was less than the heat load calculated in scenario (3).

In a letter dated September 15, 1993, the licensee revisited and re-analyzed the four scenarios using a normal batch discharge of 128 versus 116 fuel assemblies. The licensee stated that larger batch sizes may be necessary if the plant operates at a capacity factor higher than originally anticipated. A transient analysis was performed by the licensee to evaluate bulk pool temperature for each of the above scenarios. Convective heat transfer and evaporative cooling from the pool surface, and heat removal from the SFP cooling heat exchangers were credited in the analysis. The heat removal rate through the operating heat exchangers was calculated based on a proprietary thermal hydraulic computer code. To perform a conservative analysis, the heat exchangers were assumed to be fouled to their design maximum and thus the temperature effectiveness value represented the lowest postulated value calculated from heat exchanger thermal-hydraulic codes.

The most limiting scenario with regard to bulk pool temperature was found to be scenario (3), when assuming a normal discharge of 128 fuel assemblies. The calculated maximum bulk pool temperature for this scenario was found to be 164.6 °F at a time 220 hours following the reactor shutdown. Discharge scenarios (1) and (2), which the licensee presented to demonstrate compliance with NUREG-0800, 9.1.3 provisions, were found to have maximum bulk pool temperatures of 140.98 °F and 161 °F, respectively, when assuming a discharge of 128 fuel assemblies. The calculated maximum bulk temperature for scenario (4) was found to be 163.2 °F when assuming a discharge of 116 fuel assemblies.

The licensing report also evaluated the transient response of the SFP following a loss of all forced cooling. The loss of cooling was assumed to occur coincident with the maximum bulk temperature reached for each scenario. To analyze the bounding event, it is also assumed that the loss of cooling occurs coincident with the failure of the gates separating the SFP from the cask pit and the reactor cavity. This failure leaves the SFP water level approximately one foot above the top of the spent fuel racks at the time that cooling is lost. The transient response was evaluated assuming no makeup water addition. Of the scenarios evaluated, scenario (3) was found to be the most limiting regardless of the number of fuel assemblies discharged (116 or 128) due to the higher temperature at the time the cooling is lost. For this scenario, bulk boiling conditions were determined to exist in the SFP 5.5 hours following the loss of forced cooling with a boil off rate of 43.11 gpm.

In order to verify no void formation occurs and the cladding integrity is not threatened, a model was developed to calculate the maximum local water temperature and cladding temperature. The model was used to determine the location of the minimum flow in an idealized, axi-symmetric arrangement of fuel assemblies. The calculation assumed that the fuel assembly located in the minimum flow region is the most limiting. To add conservatism to the analysis, the fuel assembly cladding was assumed to have a crud deposit which covered the entire surface and all fuel assemblies were assumed to have come from the latest batch discharged simultaneously in the shortest possible time with the maximum postulated number of years of operating time in the reactor. For all scenarios with both unblocked and 50% blocked flow conditions, the calculations indicated no incidence of nucleate boiling and no potential for cladding damage.

2.1.4 Evaluation of the Thermal-Hydraulics Aspects

Section 9.1.3 of the SRP provides guidance in evaluating the heat load imposed on the SFP cooling system. The guidance specifies the evaluation of the following two scenarios: (1) a SFP inventory consisting of one normal refueling offload after 150 hours decay, one normal refueling offload after one year decay, one normal refueling offload after 400 days decay, with a single active failure of the SFP cooling system; and (2) a full core offload after 150 hours decay, one normal refueling offload after 36 days decay, and one normal refueling offload after 400 days decay where both cooling trains remain operable.

The staff compared the assumptions made by the licensee for the first scenario (normal discharge, single active cooling failure) with those stated in the SRP and found the licensee's assumptions less conservative. The staff then calculated the heat loads for the fuel inventories representative of the limiting design basis scenario for a normal discharge. The staff used the methodology of Branch Technical Position ASB-9.2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," Rev. 2, July 1981, in calculating the heat load, and assumed 4.5 years of full power operation for all stored fuel. Since the staff performed a steady state calculation, the total heat load was assumed to be constant for the scenario.

Using the calculated heat load, the heat exchanger temperature effectiveness factor calculated by the licensee, the approximate heat loss due to evaporation, and design data provided in the UFSAR, the staff calculated the SFP steady-state temperature for the normal offload scenario. The calculated maximum steady state temperature for the limiting normal offload scenarios was found to be 146 °F for a 128 fuel assembly discharge, which is higher than the 140 °F limit specified in NUREG-0800, Section 9.1.3. However, the staff finds that this value is acceptable for meeting Section 9.1.3 of the SRP because the assumptions made by the staff in conducting a steady state heat load analysis were conservative, and because the maximum pool temperature will only occur for a short duration.

The staff verified the licensee's analysis of the full core offload scenario and found the licensee's assumptions to be more conservative than those used in the SRP. Therefore, the staff used the licensee's assumptions to evaluate the full core offload scenario.

Using the assumptions from licensee scenario (3), the staff verified the calculations for the maximum SFP bulk temperature and found the maximum temperature to be 164.6 °F, which is below the temperature associated with the onset of boiling. Therefore, the guidance of Section 9.1.3 of the SRP is met with regard to providing adequate cooling for the postulated SFP inventory under full core offload conditions.

Using the assumptions from licensee scenario (3), which provide the maximum heat load to the SFP, the licensee calculated a minimum time of 5.5 hours to reach bulk boiling in the SFP following a loss of all forced cooling. Based on the availability of alternate sources of makeup water from the ESW system and from fire hoses, the staff concludes that adequate time is available to provide makeup water to the SFP prior to the onset of bulk boiling.

For the potential fuel inventory following the proposed reracking of the SFP, the cooling and makeup water supply to the SFP is adequate to meet the applicable guidance contained in Section 9.1.3 of the SRP. Therefore the staff finds the proposed reracking acceptable with regard to potential thermal-hydraulic concerns.

2.1.5 Conclusions

The staff has reviewed the licensee's submittals for the proposed SFP reracking with regard to control of heavy loads and thermal-hydraulic concerns. The staff has determined that the licensee's commitment to comply with the criteria of NUREG-0612 with regard to the control of heavy loads during the reracking operations is acceptable. The licensee's analysis demonstrated the adequacy of the SFP cooling and makeup systems in supporting the potential increased decay heat load permitted by the reracking process. The staff found this analysis to be acceptable in addressing the potential SFP thermal-hydraulic concerns.

2.2 EVALUATION OF THE CRITICALITY ASPECTS

2.2.1 Background

The proposed storage rack consists of an egg-crate structure with fixed neutron absorber material, Boral, of 0.0162 g/cm^2 boron-10 areal density, positioned between the fuel assembly storage cells. The minimum boron-10 loading in the Boral is 0.015 g/cm^2. The nominal center-to-center spacing between fuel assemblies is 6.060 inches. The 0.060-inch thick stainless-steel box, which defines the fuel assembly storage cell, has a nominal inside dimension of 5.90 inches.

The design basis fuel assembly used for the criticality analyses is a standard 8x8 array of BWR fuel rods containing UO_2 clad in Zircaloy. An initial U-235 enrichment of 4.0 weight percent (w/o) was assumed as well as 2 w/o gadolinia burnable poison (Gd_2O_3) in 8 fuel rods. The analysis was performed at the maximum fuel reactivity over burnup, which occurs at a burnup of approximately 8 MWD/KgU.

2.2.2 Evaluation

The analysis of the reactivity effects of fuel storage in the DAEC racks was performed with the two-dimensional transport theory computer code, CASMO-3. Independent verification calculations were made with the KENO-5a Monte Carlo computer code using the 27-group SCALE cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with CASMO-3. 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 parameters important to reactivity such as enrichment, assembly spacing, and absorber thickness. These two independent methods of analysis (KENO-5a and CASMO-3) showed good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 500,000 neutron histories in 1000 generations of 500 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-5a reactivity calculations. 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 authorized to be stored without any control rods or any uncontained burnable poison, and with the fuel at the burnup corresponding to the highest reactivity during its burnup history.

(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 of storage cells in all directions (except for the assessment of certain abnormal conditions where neutron leakage is inherent).

(4) Neutron absorption effect of minor structural material, such as spacer grids, is neglected.

The staff concludes that appropriately conservative assumptions were made.

The design basis reactivity calculations accounted for uncertainties due to manufacturing tolerances in boron loading, boral width, cell lattice spacing, stainless steel thickness, and fuel enrichment and density. These uncertainties were appropriately determined to at least the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a calculational bias and uncertainty were determined from benchmark calculations as well as an allowance for uncertainty in depletion calculations. The proposed design, when fully loaded with fuel enriched to 4.0 w/o U-235 with a gadolinia content of 2 w/o in 8 rods, resulted in a maximum calculated effective multiplication factor (k_{eff}) of 0.9348, when combined with all known uncertainties, including a conservative allowance for possible differences between previous fuel vendor calculations and those reported here. This meets the staff's criterion of k_{eff} no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, and is, therefore, acceptable.

The licensee also provided criticality calculations for storage of more highly enriched fuel in the SFP. The licensee considered fuel with initial average enrichments of 4.25 w/o, 4.6 w/o, and the GE-11 9x9 fuel rod array at 4.6 w/o U-235. Calculations were made for these various fuel types in both the spent fuel storage rack configuration and the DAEC core geometry (6.0-inch assembly pitch, 20 °C). An infinite multiplication factor (k_∞) of 1.31 for any of these fuel types in the standard DAEC core geometry, which is defined as an infinite array of fuel assemblies located on a 6-inch lattice spacing in unborated water at 20 °C, without any control rods or voids present, was found to result in a k_{eff} of less than 0.95 when stored in the DAEC storage racks. Therefore, for acceptable storage of fuel in the DAEC storage racks, (1) the fuel must have an average enrichment of 4.6 w/o U-235 or less, and (2) the k_∞ in the standard DAEC core geometry, calculated at the maximum over burnup, must be less than or equal to 1.31. These requirements are incorporated into DAEC Technical Specification 5.5.2, "Spent and New Fuel Storage." The licensee has also shown that any fuel of 3.1 w/o average enrichment or less is acceptable regardless of the gadolinium content or the k_∞ in the standard core geometry. These criteria are expected to bound all present and future fuel designs anticipated to be used at DAEC.

The licensee considered the reactivity effects of abnormal and accident conditions due to temperature and water density effects, abnormal or eccentric fuel assembly positioning, zirconium fuel channel distortion, fuel rack lateral movement, or the drop of a fuel assembly on top of the storage rack. None of the credible conditions resulted in exceeding the SRP maximum reactivity criterion of $k_{eff} \leq 0.95$.

2.2.3 Conclusion

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

2.3 STRUCTURAL ASPECTS

The primary purpose of this review is to assure the structural integrity of the rack modules and stored fuel assemblies, and the spent fuel pool structure under the postulated loads (Appendix D of SRP Section 3.8.4), and fuel handling accidents. There were three conference calls held on December 28, 1993, and January 6 and 7, 1994, to clarify the licensee's responses.

2.3.1 High Density Racks

The high density spent fuel storage racks are seismic Category I equipment, and are required to remain functional during and after a safe shutdown earthquake (SSE). IEL&P used a computer program, DYNARACK, for dynamic analysis to demonstrate the structural adequacy of the DAEC spent fuel rack design under earthquake loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements as defined in the program was used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two seismic analyses were performed: the 3-D single rack model analysis and the 3-D whole pool multi-rack (WPMR) analysis. The main purpose of the WPMR analysis was to investigate the fluid-structure interaction effects between racks and pool walls as well as those among the racks. These seismic analyses were performed utilizing the direct integration time-history method. Four sets of seismic time histories were calculated from the plant response spectra as described in the DAEC Updated Final Safety Analysis Report (UFSAR) (Reference 6), and each set consists of three statically independent time histories for two horizontal and the vertical directions. The average calculated response spectra generated from these time histories envelop the DAEC design response spectra.

In the 3-D single rack model analysis, three rack geometries were considered for the calculation of stresses and displacements: (1) 11 ft x 21 ft, (2) 14 ft x 21 ft, and (3) 17 ft x 19 ft. Each rack was considered fully loaded, partially loaded, and almost empty with two different coefficients of friction between the rack and the pool floor (μ=0.2 and 0.8) to identify the worst case response for rack movement and for rack member stresses and strains.

Each of the three racks was subjected to the controlling loading condition (Dead Load + Thermal Load + SSE). The calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code (1989 edition), Section III, Subsection NF. Tables 6.7.3-6.7.20 of Reference 1 present the stress factors for various rack geometries, friction and loading configurations. The stress factor is defined as the ratio of the actual developed stress to the specified Level A service limit of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. For example, the limiting value of stress factor for tension with load combination including SSE (Level D service limit) is 2.0 provided the material is such that 1.2 S_y is less than 0.7 S_u. The results show that the stress factors vary from 0.11 to 0.47, and most stress factors are below 0.30 indicating that the induced stresses in the rack due to the postulated loading conditions are very small when they are compared to the allowable stresses of the ASME Code. These low stress factors indicate significant conservatism in the rack design, therefore, are acceptable.

Tables 6.7.3-6.7.20 of Reference 1 also show the calculated horizontal displacements at the top and baseplate levels of the rack. The displacements at the baseplate level and at the top level are about 0.01 inch and 0.15 inch, respectively. These computed horizontal rack displacements show that rack-to-rack impacts and rack-to-wall impacts would not occur during a SSE event.

IEL&P also calculated the weld stresses of the rack under the SSE loading condition. Three weld locations were considered: (1) baseplate-to-rack, (2) baseplate-to-pedestal, and (3) cell-to-cell connections. Table 6.7.27 of Reference 1 shows the ratio of the calculated weld stress with respect to the allowable stress specified in ASME Code Section III, Subsection NF. The calculated ratios are in the range of 0.23 to 0.31 indicating that the weld connection design of the rack is adequate and acceptable.

In the 3-D whole pool multi-rack analysis, all eleven racks were modeled together with the pool structure. All racks were considered fully loaded with one coefficient of friction (μ=0.5) between the racks and the pool floor, and they were subjected to a loading condition of (Dead Load + Thermal Load + SSE). The results of the multi-rack analysis indicate that the calculated stresses of a rack are higher than those obtained from the corresponding single rack analysis. However, all calculated stresses of the multi-rack analysis are smaller than the allowable stress of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, and are acceptable. The results of the multi-rack analysis also show that no rack-to-wall or rack-to-rack impact is expected.

Based on: (1) the IEL&P's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) large factor of safety of the induced stresses and strains of the rack when they are compared to the corresponding allowables provided in the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, and (3) IEL&P's overall structural integrity conclusions supported by both single and multi-rack analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions, and therefore, are acceptable.

However, it is quite likely that the racks will move during or after seismic events. Therefore, IEL&P is required to institute a surveillance program that inspects and maintains the originally installed rack gaps after occurrence of an earthquake equivalent to or larger than an Operating Basis Earthquake (OBE), if any occurs.

2.3.2 Spent Fuel Storage Pool

The spent fuel pool structure is a reinforced concrete structure and is designed as a seismic Category I structure. The dimension of the DAEC pool structure is approximately 24 feet wide and 40 feet long with 6 foot thick reinforced concrete slab. The internal surface of the pool structure is lined with stainless steel to ensure water tight integrity.

The pool structure was analyzed by using the finite element computer program, ANSYS, to demonstrate the adequacy of the pool structure under fully loaded high density fuel racks with all storage locations occupied by fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in SRP Section 3.8.4 including the thermal loads.

Table 8.5.1 of Reference 1 shows the predicted factors of safety varying from 1.44 to 4.40 for bending moments of concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the IEL&P pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

2.3.3 Fuel Handling Accident

The following two accident cases were evaluated by IEL&P: (1) drop of a fuel assembly with handling equipment, which enters an empty cell and impacts the baseplate, and (2) drop of a fuel assembly with a channel, which impacts the top of a rack.

The analysis results of Accident Case (1) above shows that the load transmitted to the liner through the structure is properly distributed through the bearing pads located near the fuel handling area, thereby, the liner is not damaged by impact. The analysis results of Accident Drop Case (2) shows that damage will be restricted to a depth of 1.09" below the top of the rack, which is above the active fuel region.

The staff reviewed the IEL&P analysis results submitted, and concurs with its findings.

2.3.4 Conclusion

Based on the review and evaluation of IEL&P's March 23, 1993, and supplemental submittals with additional information and analysis, it is concluded that the IEL&P's structural analysis and design of the spent fuel rack modules and the spent fuel pool structure are adequate to withstand the effects of the required loads. The analysis and design are in compliance with current licensing basis set forth in the UFSAR and applicable provisions of the Standard Review Plan (SRP), therefore, are acceptable provided that IEL&P commits to implement a surveillance program that inspects and maintains the originally installed rack gaps after occurrence of an earthquake equivalent to or larger than an OBE.

2.4 EVALUATION OF THE RADIATION PROTECTION ASPECTS

2.4.1 Occupational Dose

The licensee estimated in its March 26, 1993, application that total occupational dose for planned reracking operations would be between 6 and 12 person-rem, including any necessary diving activities.

This overall estimate is based on individual dose estimates for each of the series anticipated activities to be performed during the reracking operation. These activities including removing and decontaminating (hydrolasing) the current racks once they are emptied and removed from the fuel pool; removing underwater appurtenances; installing new racks; and preparing the old racks for shipping.

The licensee has indicated that the underwater appurtenances will be removed using remote handling tools to the greatest extent possible. If diving operations are required, careful monitoring and adherence to procedures should ensure that the radiation dose to the divers is as low as reasonably achievable (ALARA). Further, if divers are used, the licensee has substantially committed to the guidance provided in Appendix A ("Procedures for Diving Operations in High and Very High Radiation Areas") to Regulatory Guide 8.38 (Ref. 5).

The radiation protection program at the DAEC is adequate for the reracking operations. Where there is a potential for significant airborne activity, continuous air samplers will be in operation. Personnel will wear protective clothing and, if necessary will use, respiratory protective equipment. Work activities will be governed by a radiation work permit, and personnel-monitoring equipment will be issued to each individual as needed. As a minimum, this will include thermoluminescence dosimeters and pocket dosimeters. Additional personnel-monitoring equipment (i.e., extremity badges or alarming dosimeters) may be utilized as required. All work activities, personnel traffic, and the movement of equipment will be monitored and controlled to minimize contamination and to ensure that exposures are maintained ALARA.

Based on our review of the licensee's application, the staff finds the proposed radiation protection aspects of the spent fuel pool reracking activity acceptable.

2.4.2 Solid Radioactive Waste

The licensee stated that the spent fuel storage racks will be removed and washed in preparation for packaging and shipment. Estimates of the person-rem exposures associated with this operation were included and found to be acceptable. Shipping containers and procedures will conform to DOT regulations and to the requirements of the state through which the shipment may pass, as determined by the State DOT office.

On the basis of its review, the staff finds that the licensee's plan for handling and disposing of solid radioactive waste generated during the planned reracking operation meets regulatory requirements and is, therefore, acceptable.

2.4.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated accidents, including means for avoiding them in the design and operation of the facility, and recommended means for mitigating their consequences should they occur. The licensee has evaluated the effect of the changes on the calculated consequences of a spectrum of postulated design basis accidents (i.e; fuel handling accidents and spent fuel cask drop accidents) and concludes that the effect of the proposed TS change is small and that the calculated consequences are within regulatory requirements and staff guidelines on dose values. The addition of poison pins or removal of blocking devices will not have any effect on the probability of occurrence of either of these two accidents. Since the licensee proposes to utilize extended burnup fuel, the staff reevaluated the fuel handling accident for the DAEC to consider the effect of increased burnup.

In its evaluation for DAEC, issued on January 23, 1973, the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in Standard Review Plan (SRP) Section 15.7.4.

Since the licensee intends to utilize extended burnup fuel, the staff reanalyzed the fuel handling DBA for this case. According to NUREG/CR-5009 (Ref. 4) increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T increases the doses for a fuel handling accident by a factor of 1.2. The licensee proposes to increase fuel enrichment to 4.6 weight percent U-235 with a maximum burnup of 60,000 MWD/T. The 1.2 factor increase in dose displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guideline doses in SRP Section 15.7.4 (established on the basis of 10 CFR Part 100).

Table 1

Radiological Consequences of Fuel Handling Design Basis Accident (rem)

	Thyroid	
	Exclusion Area	Low Population Zone
Staff Evaluation January 23, 1973	< 1	< 1
Bounding Estimates for Extended Burnup Fuel[1]	1.2	1.2
Regulatory Requirement (NUREG-0800 Section 15.7.4)	75	75

The staff concludes that the only potential increased doses resulting from the fuel handling accidents with extended burnup fuel is the thyroid doses; these doses remain well within the dose limits given in NUREG-0800 and are, therefore, acceptable.

2.5 PROPOSED TECHNICAL SPECIFICATION REVISIONS

Technical specification page 5.5-1 was revised based on the limitations necessary to maintain stored fuel k_{eff} less than or equal to 0.95. Also editorial changes were made to correct a reference and to clarify k_{eff}. References were added to this page for the Holtec report and the GE Fuel Bundle Designs report. These changes resulted in the deletion of page 5.5-2.

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[1]Factor of 1.2 greater than original estimate for iodine.

Page 3 of the Operating License was administratively changed to reflect the Updated Final Safety Analysis Report as well as the letter of March 26, 1993, requesting this amendment change. The staff finds these changes acceptable.

2.6 SUPPLEMENTAL INFORMATION

The supplemental information in letters dated September 15, November 23, 1993, and January 10, 1994, were either administrative or editorial in nature to clarify the original request and did not contain substantive changes to the original submittal.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATIONS

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on January 4, 1994 (59 FR 786). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J Minns
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Date: February 2, 1994

REFERENCES:

1. Application for Amendment to License No. DPR-49, revising Technical Specification 5.5, dated March 26, 1993.

2. NUREG-0800, Section 12.6, "Occupational Exposure Associated with the Spent Fuel Pool," November 1984.

3. Summary of Meeting held July 8, 1993, regarding DAEC Spent Fuel Pool Reracking Amendment Request, dated July 15, 1993.

4. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," PNL, 1987.

5. Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," June 1993.

6. Duane Arnold Energy Center, Updated Final Safety Analysis Report.

7. IEL&P, Letter dated November 23, 1993, "Duane Arnold Energy Center, Docket No. 50-331, Op. License No: DPR-49, Response to Second Request for Additional Information on Spent Fuel Pool Rerack Licensee Amendment Request."

UNITED STATES NUCLEAR REGULATORY COMMISSION

IOWA ELECTRIC LIGHT AND POWER COMPANY, ET AL.

DOCKET NO. 50-331

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 195 to Facility Operating License No. DPR-49, issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and the Corn Belt Power Cooperative., which revised the Technical Specifications for operation of the Duane Arnold Energy Center located in Linn County, Iowa. The amendment was effective as of the date of issuance.

The amendment revised Technical Specification 5.5 based on the limitations necessary to maintain stored fuel k_{eff} less than or equal to 0.95 for the reracking of the spent fuel pool. The revision also made some editorial changes and added two references. Page 3 of the Facility Operating License was revised to reflect the Updated Final Safety Analysis Report and the letter of March 23, 1993, requesting this amendment change.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the Federal Register on July 30, 1993 (58 FR 40841). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an Environmental Impact Statement. Based upon the Environmental Assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action see (1) the application for amendment dated March 23, 1993, and supplemented September 12 and November 23, 1993, and January 10, 1994; (2) Amendment No. 195 to License No. DPR-49, (3) the Commission's related Safety Evaluation dated February 2, 1994 , and (4) Environmental Assessment dated January 6, 1994 (59 FR 786). All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building, 2120 L Street NW., Washington, DC, and at the Cedar Rapids Public Library, 500 1st Street, S.E., Cedar Rapids, Iowa 52401.

Dated at Rockville, Maryland this 2nd day of February 1994

FOR THE NUCLEAR REGULATORY COMMISSION
Original signed by

Robert M. Pulsifer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Concurred on other copy 1/31/93. See other copy for changes APH 2/01/94

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