February 24, 1993

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Docket No. 50-373

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M83697)

The Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. NPF-11 for the LaSalle County Station, Unit 1. The amendment is in response to your application dated June 5, 1992, as supplemented July 7, July 20, and November 4, 1992.

The amendment revises Technical Specification 5.6, "Fuel Storage," to permit the storage of up to 3986 fuel assemblies in the Unit 1 spent fuel pool.

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

Sincerely,

Original signed by
John F. Stang for:

Robert J. Stransky, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 90 to NPF-11
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PDIII-2	PM:PDIII-2	OGC	D:PDIII-2
NAME	CMOORE	BSIEGEL		JDYER
DATE	2/19/93	2/16/93	2/17/93	2/24/93

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

February 24, 1993

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John F. Stang FOR
Robert J. Stransky, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 90 to NPF-11
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

LaSalle County Station
Unit Nos. 1 and 2

cc:

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated June 5, 1992, as supplemented July 7, July 20, and November 4, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 90, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following page of the Appendix "A" Technical Specifications with the enclosed page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
5-5	5-5

5.6 FUEL STORAGE

CRITICALITY

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

a. A k_{eff} equivalent to ≤ 0.95 when flooded with unborated water, including all calculational uncertainties and biases, as described in Section 9.1 of the FSAR.

b. A nominal 6.26 inch center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 819 feet.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 3986 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7.1-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. NPF-11

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNIT 1

DOCKET NO. 50-373

1.0 INTRODUCTION

By letter dated June 5, 1992, the Commonwealth Edison Company (CECo, the licensee) proposed to amend the Technical Specifications (TS) for the LaSalle County Station, Unit 1, to permit the use of high density spent fuel racks. The LaSalle County Station, Unit 1, received a full power operating license on August 13, 1982. At the time of licensing, the Unit 1 spent fuel pool (SFP) contained sufficient storage capacity to accommodate 1080 fuel assemblies. The new proposed high-density storage racks would increase the storage capacity of the Unit 1 SFP to 3986 fuel assemblies.

The LaSalle County Station is equipped with two SFPs; one associated with Unit 1 and one associated with Unit 2. The licensee has the capability to transfer fuel between the two SFPs. On June 15, 1989, the Commission authorized the licensee to increase the storage capacity of the Unit 2 SFP to 4078 fuel assemblies. The licensee has estimated that sufficient capacity exists in the current SFPs to provide storage capacity until the year 2002, while maintaining full core offload capability for both units. The proposed installation of high-density spent fuel racks in the Unit 1 SFP is projected to provide adequate storage capacity until the year 2013, while maintaining the full-core offload capability of both units.

2.0 EVALUATION

The NRC staff's review of the licensee's proposed amendment is based upon the licensee's June 5, 1992, submittal, as well as supplemental information provided by the licensee in letters dated July 7, July 20, and November 4, 1992. The licensee's submittals were reviewed for compliance with the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2, "Design Bases for Protection From Natural Phenomena," GDC 44, "Cooling Water," and GDC 61, "Fuel Storage and Handling and Radioactivity Control," as well as NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and applicable sections of NUREG-0800, "Standard Review Plan." The July 7, July 20, and November 4, 1992, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.1 Decay Heat Generation Rate

The licensee evaluated the impact of two specific discharge cases for full core offloads into the SFP. In both cases, the licensee assumed that a normal refueling batch (256 fuel assemblies) had been deposited into the SFP. This is assumed to be the thirteenth refueling batch for a total of 3328 assemblies in the SFP. After a refueling period of 45 days the plant is assumed to be restarted. Case 1 assumed that the full core is unloaded after 30 days of reactor operation; Case 2 assumed that the core is unloaded after 60 days of operation. In both cases, the total content of the SFP is assumed to be 4092 spent fuel assemblies (SFA), which is larger than the actual maximum storage capability of 3986 assemblies.

The licensee provided no cases directly to show the SFP coolant conditions for the condition of a "normal" refueling offload of 256 SFAs with one pump and one heat exchanger (HX), i.e., one "train" in operation. However, the licensee suggested the use of such previous calculations made for the Unit 2 SFP for reracking. This is discussed below.

The licensee reported total heat generation loads of 37,000,000 BTU/HR for Case 1 and 37,800,000 BTU/HR for Case 2; these values apply at the time the coolant attains its maximum temperature, 180 hours after shutdown. The licensee also reported that the Updated Final Safety Analysis Report (UFSAR) value of 42,000,000 BTU/HR which was applicable for the Unit 2 reracking serves to act as a bound for the Unit 1 value. The staff agrees with the value of 42,000,000 BTU/HR for the full core offload condition and with the values for Cases 1 and 2. As for the normal offload condition, the staff finds agreement with the value of 17,600,000 BTU/HR calculated previously by the staff for the Unit 2 rerack.

The calculation of decay heat generated is based upon the licensee's assumption that fuel elements are loaded into the SFP at the rate of 10 per hour, starting at 100 hours after reactor shutdown. However, the plant TS allow fuel assemblies to be removed from the core, starting at 24 hours after shutdown. The licensee agreed to include the 100 hour delay in unloading procedures, in order to assure compliance with calculated values of decay heat generation in the SFP.

2.2 Spent Fuel Pool Cooling System

The SFP cooling, filter, and demineralizer system for each unit contains two cooling pumps, two HXs and two filter/demineralizers (F/Ds). The two pumps are arranged in parallel, as are the HXs and F/Ds. In this way, it is possible for each pump to operate with either HX and either F/D. The system is arranged so that flow from the pumps is directed first to the F/Ds and then to the HXs. When necessary, the F/Ds may be bypassed with the SFP coolant passing directly to the HXs and then returning to the pool.

2.2.1 Maximum Spent Fuel Pool Bulk Coolant Temperature

The licensee used a transient calculation to determine the maximum SFP bulk coolant temperatures for full core offload Cases 1 and 2. The maximum bulk pool temperatures for the two offload cases were found to be 126.5 and 127.2 °F, respectively, assuming full flow (6000 gpm) through the SFP cooling system. The licensee initially reported, however, that their calculations showed coolant flow through the SFP cooling system, using both pumps, both F/Ds, and both HXs to be 5050 gpm, in contrast to the design value of 6000 gpm, and estimated that the actual bulk coolant temperatures would be less than 12 °F greater than those calculated using the design specifications. The licensee subsequently calculated the difference to be 1.4 °F (128.6 °F as compared with the original Case 2 value of 127.2 °F).

The staff, in analyzing the cooling capability of the Unit 2 SFP cooling system with one train operating, determined that the SFP bulk coolant temperature would be less than 126 °F during a normal refueling (assuming the maximum normal heat load) with one set of SFP cooling system components in operation, for a SFP filled with 4078 assemblies (larger than the actual maximum pool capacity of 3986 assemblies).

2.2.2 Protection of Filter/Demineralizer Resins

The licensee reported that the demineralizer resin could be expected to degrade or be damaged by SFP coolant temperatures in excess of 140 °F. However, the temperature of the coolant passing into the demineralizer during operation of maximum normal heat load (normal offload) with one pump, and maximum abnormal heat load (full core offload) with two pumps operating, has been calculated to be less than 130 °F. Only in the unlikely event of a full core offload combined with failure of one pump and at least one HX is there expected to be a problem. In such case, the F/D system could be bypassed without reducing the cooling capability of the remaining components.

2.2.3 Maximum Fuel Cladding Temperature

The licensee calculated the maximum fuel cladding temperature by first calculating the maximum local water temperature in the pool for the case of a full core offload with both SFP cooling "trains" operating (Case 2). The maximum cladding temperature was then determined by using the axial peaking factor in two calculations: one in which it was assumed that 50% of the fuel cell was blocked; and one case in which the cell was unblocked. Blockage was assumed to result from a misplaced fuel assembly. The maximum cladding temperatures were determined to be 215 °F and 211 °F, for the first and second cases, respectively.

Other conservatisms in the licensee's calculation included: (1) the use of a crud deposit on the cladding equivalent to a resistance of 0.005 $°F\text{-}FT^2\text{-}hr/BTU$; (2) the use of the average fuel element operating power in lieu of the power after a period of decay; and (3) the use of an idealized, conservative flow geometry.

2.2.4 Spent Fuel Pool Coolant Time to Boil

In calculating the time required for the pool coolant to boil, the licensee assumed that the plant lost the use of the SFP cooling system at the time the coolant attained its maximum temperature during the introduction of spent fuel from the reactor core. The results for the cases studied are as described in Table 1.

TABLE 1
Spent Fuel Pool Time To Boil
Assuming No Coolant Addition (hours)

Offload Case	Time To Boil	Maximum Evaporation Rate
1a[1]	6.10	78.88 gpm
1b[2]	6.12	78.43
2a[1]	5.91	80.65
2b[2]	5.93	80.19

[1]Cooling loss occurs at point of maximum decay heat generation

[2]Cooling loss occurs at point of maximum temperature

2.2.5 Alternate Cooling Methods

2.2.5.1 Use of Residual Heat Removal System

In the event the SFP cooling system is completely inoperable, the B train of the residual heat removal (RHR) system may be employed to cool the SFP once the full core has been unloaded into the pool.

2.2.5.2 Coolant Addition

In the event conditions occur which allow the SFP coolant to boil, normal makeup may be made from the cycled condensate storage system. This system, however, is not safety-related and may not be available, except during normal operating conditions. In the event that the cycled condensate storage system is not available, SFP coolant makeup may be provided by the core standby cooling water system - equipment water cooling system (CSCS-EWCS), which is intended as an emergency makeup water source for the SFP. The CSCS-EWCS is safety-related and meets the single-failure criterion.

The Standard Review Plan (SRP) acceptance guidelines require that the SFP coolant temperatures be maintained at 140 °F or less when the SFP is filled with SFAs during and following a normal core reload, with one cooling train operating. The SRP also requires that the SFP coolant remain below the bulk boiling temperature of 212 °F with the SFP filled after a full core off-load. The proposed SFP configuration conforms to these guidelines. The demineralizer resin is protected against damage during normal operation with one cooling train in operation, and even in the event of a full core offload with two trains in operation. The calculated spent fuel cladding temperature, with all the conservative assumptions applied, is calculated to remain below the nominal temperature experienced during reactor operation, and is, therefore, acceptable. The staff finds that the bulk coolant time to boil is acceptable since it provides a reasonable time to allow operators to use alternative methods to cool the SFP coolant, or to provide makeup coolant in the event that the SFP coolant begins to boil.

In view of the foregoing information, the staff finds that all SFP cooling concerns related to the proposed reracking have been adequately addressed.

2.3 Heavy Load Handling

2.3.1 Reactor Building Crane

The licensee proposes to use the reactor building crane to move old racks out of the pool and new racks into the pool. The reactor building crane has a main hoist which is designed to lift a load of 125 tons and has been tested to 125% of its rated capacity (156.25 tons). The main hoist cables have a factor of safety of 600% or greater and have a combined breaking strength of at least 750 tons. Since the heaviest rack, combined with the special lifting device and 15 ton auxiliary crane hoist, weigh approximately 17 tons, the hoist cables would exhibit a factor of safety of 44 when using the crane to lift the new racks. The factor of safety is an approximate indication of the crane capacity before failure.

2.3.2 Special Lifting Device

A special lifting device (rig) has been designed for use in moving the existing and new racks with the reactor building crane. The device has four legs (lift rods) which fit into a square section; the square section of the rig is not designed to carry any loads, but merely to locate the legs of the rig in the proper position for lifting. Four lift eyes, one at each corner of the square section of the rig, will connect to four independent cables and transmit the load from the legs of the rig to the crane. The licensee plans to align the center of gravity of the load to be lifted with the center of lift in order to limit any lateral motion of the rig and rack in the event of failure of one of the legs or cables.

The licensee stated that the lifting rig is designed to comply with the duality feature called for in Section 5.1.6(1) of NUREG-0612, and would not result in uncontrolled lowering of the load in the event of failure of one of the load-bearing legs. Section 5.1.6(1) of NUREG-0612, requires that the lifting device be able to carry six times the maximum load and ten times the maximum load, before reaching the yield stress or ultimate stress, respectively, of the weakest component(s). While the licensee did not provide details of the load carrying capacity of the rig, the staff noted that a comparable design at another plant was capable of hoisting 180 tons before reaching the yield stress and 300 tons before reaching the ultimate stress of the weakest component(s). This capability would translate to factors of safety of 13.4 and 22.4 to yield and ultimate stresses, thus complying with the requirements of NUREG-0612. (The maximum load carried by the special lifting device is the heaviest rack, which weighs 13.4 tons.)

2.3.3 Auxiliary Hoist

A 15 ton auxiliary hoist will be used between the special lifting device and the reactor building crane in order to prevent the crane hook from being immersed in the SFP coolant during the reracking process. The licensee stated that the purchase specification for this hoist required that it be single-failure-proof.

2.3.4 Other Lifting Equipment

Turnbuckles will be used to minimize horizontal movement when lifting or lowering racks. The licensee will also align the lift point with the center of gravity of the load being lifted in order to minimize sway in the event of a lifting rig leg failure.

2.3.5 Load Paths

The licensee has developed safe load paths for the movement of all racks within the reactor building. In this manner, redundant safe shutdown equipment or spent fuel will not be affected in the event of a load drop.

2.3.6 Other Heavy Load Considerations

The licensee has committed to perform other actions to improve safety during the reracking process. These are: (1) to give the crane and hoist a preventative maintenance checkup and inspection prior to the start of reracking; (2) to train all crew members involved in reracking in the use of the lifting and upending equipment; and (3) to provide operating procedures covering the operations involved in reracking.

The licensee's heavy loads handling concerns are resolved in that the handling train, consisting of the reactor building crane, 15 ton auxiliary hoist, and special lifting device, may be considered single-failure-proof as noted above. Applicable guidelines assume that use of a single-failure-proof handling train assures that the potential for a load drop is extremely small and satisfies all heavy load handling concerns.

In addition, the licensee complies with alternative guidelines requiring compliance with the following evaluation criteria assuming the drop of a heavy load during the reracking procedure:

(1) Release of radioactive material:

There is no potential for release because there will be no spent fuel in the Unit 1 SFP or in the path of heavy loads movement during the reracking process.

(2) Damage to fuel and racks so as to affect criticality:

Criticality will not be affected since heavy loads cannot be dropped on spent fuel as discussed in (1) above.

(3) Damage to the SFP:

The licensee has analyzed the effect of a dropped rack and stated that "the maximum load due to the rack drop event is well below the cumulative impact load produced during the seismic event," which the licensee claims to be acceptable based upon the maximum calculated SFP wall and floor stresses. This assertion has not been reviewed in detail by the staff because of the license's use of the single-failure-proof handling train. However, previous analyses of drops of similar racks have shown that SFPs suffer only minor damage due to the drop of a rack.

(4) Damage to redundant safe shutdown equipment:

The licensee has stated that pathways for moving heavy loads will not be such as to endanger redundant safe shutdown equipment in the event of a load drop; i.e., the licensee will use safe load paths.

Based upon our review of the licensee's submittals, the staff finds the heavy load handling portion of this review to be acceptable.

2.4 <u>Criticality Evaluation</u>

The proposed storage rack cell consists of an egg-crate structure with fixed neutron absorber material, Boral™, of 0.0238 g/cm^2 boron-10 areal density positioned between the fuel assembly storage cells. The nominal center-to-center spacing between fuel assemblies is 6.264 inches. The 0.090 inch stainless steel box which defines the fuel assembly storage cell has a nominal inside dimension of 6.05 inches.

The design-basis fuel for the storage racks is an 8x8 Boiling Water Reactor (BWR) fuel rod assembly with an average fuel enrichment of 3.5 weight percent (w/o) uranium-235 (^{235}U), including the 6-inch long natural UO_2 blankets located at the uppermost and lowermost six inches of the fuel rods. The uniform average enrichment in the 138 inch long enriched zone of the fuel rods is 3.743 w/o ^{235}U.

The analysis of the reactivity effects of fuel storage was performed with both the CASMO-3 computer code (a two-dimensional multi-group transport theory code) and KENO-5a (a Monte Carlo code), using the 27 energy group SCALE neutron cross-section library. CASMO-3 was also used as a means of evaluating small reactivity increments associated with manufacturing tolerances. These codes are widely used for the analysis of fuel storage rack reactivity, and have been benchmarked against results from numerous criticality experiments. The staff concludes that the analytical methods used are acceptable.

The criticality analyses were performed using several assumptions which would tend to maximize the reactivity of the spent fuel rack. These include:

(1) A uniform average ^{235}U enrichment of 3.75 w/o (compared to an actual 3.743 w/o ^{235}U) in the enriched zone of the fuel, with no credit given for gadolinium burnable absorber.

(2) Unborated pool water at the temperature yielding the maximum reactivity (4° C).

(3) Assumption of an infinite array of storage cells in the lateral direction.

(4) Neutron absorption by the structural material of the racks and fuel 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 density and ^{235}U enrichment. These uncertainties were appropriately determined to at least the 95 percent probability, 95 percent confidence (95/95) level. In addition, the calculational bias and uncertainty were determined from benchmark calculations. The proposed design, when fully loaded with fuel enriched to 3.75 w/o ^{235}U, resulted in a calculated effective neutron multiplication constant (k_{eff}) of 0.943 when analyzed using the CASMO-3 code and combined with all known uncertainties. Independent calculations using the KENO-5a code resulted in a k_{eff} of 0.942, which agrees well with the CASMO-3 result. These results meet the acceptance criterion of maximum $k_{eff} \leq 0.95$, including all uncertainties at the 95/95 probability/confidence level. The staff concludes that these calculations are 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 up to 4.25 w/o ^{235}U (4.6 w/o in the enriched zone) with 8 rods per assembly containing 2 w/o gadolinium (as Gd_2O_3). Gadolinium and uranium depletion calculations were performed using CASMO-3 to determine the variations in fuel reactivity as a function of fuel burnup. The licensee determined that higher enrichments, with the gadolinium loading commonly used in BWR fuel, resulted in pool reactivities which were considerably lower than those calculated for the normal 3.75 w/o enriched fuel without gadolinium. The peak fuel pool reactivity calculated for 4.6 w/o enriched fuel was 0.8973, which occurred at a burnup of 9 megawatt-days per kilogram of uranium (MWD/kgU), and is well below the acceptance criterion of $k_{eff} \leq 0.95$. This corresponds to an infinite neutron multiplication constant (k_∞) of 1.332 in the standard LaSalle core geometry, which is defined as an infinite array of fuel assemblies located in a 6-inch lattice spacing in unborated water at 20 °C, without any control absorber or voids present. Thus, any fuel assembly which has a k_∞ of 1.332 or less at 20 °C in the standard core geometry, and an average enrichment of 4.6 w/o or less in the enriched zone, will result in a k_{eff} of less than 0.8973 (0.9241 including uncertainties) when stored in the spent fuel rack, and will meet the fuel storage reactivity criterion specified in the SRP. The staff considers storage of fuel with average enrichments of up to 4.25 w/o ^{235}U (4.6 w/o in the enriched zone) to be acceptable, provided that the storage configuration of fuel assemblies does not result in a local k_∞ of 1.332 or greater, when calculated as described above. The licensee will be expected to verify that the k_∞ of the limiting lattice, at all potential fuel burnups, remains less than this limit.

The licensee considered the reactivity effects of abnormal and accident conditions due to temperature and water density effects, eccentric fuel assembly positioning, 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 $k_{eff} \leq 0.95$.

The proposed storage racks will also contain four large, square cells with an inside dimension of 11.5 inches. These cells are designed to store control rod guide tubes or defective fuel containers. The licensee's calculations determined that the storage of defective fuel containers loaded with fresh fuel in these cells would result in a maximum k_{eff} of approximately 0.74. This result is also well within the SRP maximum reactivity requirement.

2.5 Structural Design

2.5.1 High Density Racks

The proposed 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). The licensee used a computer program, DYNARACK, for dynamic analysis to demonstrate the structural adequacy of the spent fuel rack design under earthquake loading conditions. The proposed spent fuel racks are free-standing and self-supporting equipment, and are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, spring, gap, and friction elements as defined in the program was used to simulate three dimensional dynamic behavior of the rack 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.

The seismic analysis was performed utilizing the time-history method. The seismic time histories were calculated from the plant floor response spectra (FRS) as described in the LaSalle County Station UFSAR. For stress and displacement analysis, three rack geometries were considered: (1) 15 feet x 17 feet, (2) 15 feet x 18 feet, and (3) 9 feet x 18 feet. Each rack was considered fully loaded, partially loaded, and almost empty with three different coefficients of friction between the rack and the pool floor

(μ=0.2, 0.5 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 Level C service 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 allowable stresses specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section III, Subsection NF. Tables 6.7.3 through 6.7.29 of the licensee's June 5, 1992, submittal present the stress factors for various rack geometric, friction and loading configurations. The stress factor is defined as the ratio of the calculated stress with respect to the allowable stress of Subsection NF of the Code. The limiting value of each stress factor is 1.0 for Level B service limits (Dead Load + Thermal Load + OBE) and 2.0 for Level C service limits. The results of the licensee's calculations show that the stress factor varies from 0.01 (minimum) to 0.74 (maximum) for both Level B and C service limits, and most stress factors are below 0.50, indicating that the induced stresses in the racks due to the postulated loading conditions are very small when they are compared to the allowable stresses of the ASME Code.

The licensee's June 5, 1992, submittal also provided the calculated horizontal displacements at the top and baseplate levels of the racks. The displacement at the baseplate level is less than 0.20 inch and the maximum displacement at the top level is approximately 0.22 inches. Based upon these computed horizontal rack displacements, no rack-to-wall impacts would occur during a SSE event, and rack-to-rack impact loads during the SSE would be minimal.

Commonwealth Edison Company also calculated the weld stresses of the rack under Level C service loading conditions. Three weld locations were considered: (1) baseplate-to-rack, (2) baseplate-to-pedestal, and (3) cell-to-cell connections. Table 6.7.39 of the licensee's June 5, 1992, submittal shows the ratio of the calculated weld stresses to the allowable stresses specified in Subsection NF of the ASME Code. The calculated ratios are in the range of 0.29 to 0.83, indicating that the weld connection design of the racks is adequate and acceptable.

Although CECo presented the structural design adequacy of the spent fuel rack by demonstrating that the racks would experience only small induced stresses and displacements, with no potential for overturning, during SSE loading conditions, the staff performed an independent assessment of the safety margin for overturning of a rack in order to supplement the findings obtained from the licensee's DYNARACK analysis. The assessment was based on the principle of energy conservation, whereby the kinetic energy resulting from the maximum velocity of the rack induced by a SSE loading is equated to the potential energy that is needed to raise the rack to a position where the center of gravity of the rack is about to move beyond the line connecting the two supporting legs of the rack. A conservative factor of safety is defined as the ratio of the potential energy needed to raise the rack to the point of tipping over with respect to the kinetic energy imparted to the rack by the SSE. The hydrodynamic effect is not considered in the analysis.

The staff chose a rack geometry of 9 feet x 18 feet for use in the overturning analysis, since this geometry has the narrowest width among the three rack geometries used in the licensee's analysis. The staff determined that a factor of safety of approximately 3.6 resulted when a fully loaded rack was considered. This calculated factor of safety is larger than the 1.1 required by SRP Section 3.8.5, and indicates that the rack would not overturn under a SSE loading condition.

Based on: (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction from 0.2 to 0.8, and using different geometries and loading conditions of the rack), (2) the conservatism incorporated in the analysis by neglecting the hydrodynamic effects between racks, (3) the large factor of safety between the induced stresses and displacements of the rack when they are compared to the allowable stresses provided in the ASME Code, Section III, Subsection NF, and (4) the staff's independent assessment based on simplistic but conservative assumptions, the staff concludes that the rack modules will maintain their functionality and structural integrity under postulated loading conditions, and are acceptable.

However, it is quite likely that the racks will move during or after seismic events. Therefore, CECo is required to institute a surveillance program that inspects and maintains rack gaps after an earthquake equivalent to or larger than an OBE, if any occurs. Commonwealth Edison Company should assure that the racks are in the required positions after seismic events.

2.5.2 Spent Fuel Storage Pool

The SFPs structure is a reinforced concrete structure and is designed as a Seismic Category I structure. The dimensions of the LaSalle Unit 1 pool structure are approximately 34 feet wide and 40 feet long, with a 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 SAP90 finite element computer code to demonstrate the adequacy of the pool structure with fully loaded high density fuel racks with all storage locations occupied with fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in SRP Section 3.8.4, including thermal loads. Commonwealth Edison Company identified the five critical locations of the fuel pool slab and wall sections adjoining the pool slab, based on the dynamic analysis. The results of the SAP90 structural dynamic analysis were used as input to a static computer program, which calculates reinforcing steel and concrete stresses in accordance with the requirements of the American Concrete Institute (ACI) Code (Reference 5).

Tables 8.1 and 8.2 of the licensee's June 5, 1992, submittal show the factors of safety for bending moments and shear forces, respectively, at critical locations of the pool structure. The factors of safety vary from 1.04 to 2.28 for bending moments, and from 1.25 to 4.13 for shear forces at different critical locations. The staff concludes that these factors of safety are acceptable.

In view of the calculated factors of safety, the staff concludes that the CECo fuel pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading, and SSE loading conditions. Thus, the fuel pool design is acceptable as presented.

2.5.3 Fuel Handling Accident

The following five fuel handling accident cases were evaluated by the licensee:

(1) A fuel assembly is dropped from a height of 30 inches above a storage location and impacts the base of the module.

(2) A fuel assembly is dropped from a height of 30 inches above the rack and falls straight down, hitting the top of the rack.

(3) A fuel assembly is dropped from a height of 30 inches above the rack and falls at an incline, impacting the top of the rack with both horizontal and vertical forces.

(4) Tools and equipment are dropped over a region of storage cells.

(5) An uplifting load of 1200 pounds is imposed on one storage cell.

The licensee's analysis of drop case (1) above shows that the load transmitted to the fuel pool liner through the rack supports is properly distributed through the bearing pads located on the liner and that the liner would not be damaged by the impact. The results of the licensee's analysis of drop case (2) shows that damage to the rack would be restricted to a depth of 3.2 inches below the top of the rack, which is above the active fuel region.

The licensee's analysis of drop cases (3) and (4) indicates that the results of case (2) above bound these cases, and that damage to the rack would be confined to the regions above the active fuel. Finally, the analysis of case (5) demonstrates that the stresses induced in the rack by this load are bounded by those stresses induced by the other postulated fuel handling accidents.

2.6 Radiological Considerations

2.6.1 Occupational Exposure Controls

The proposed rerack of the Unit 1 SFP will increase the storage capacity from 1080 fuel storage cells to 3986 spent fuel storage locations, plus 43 special storage cells. The licensee has stated that the storage of a full core of spent fuel (with three years of burnup at full power) in a 28 x 28 storage matrix next to the pool walls will result in a dose rate below the floor of the pool of less than 1 mrem/hr. Dose rates adjacent to the pool outer walls will vary from 1 mrem/hr to 2.5 mrem/hr, depending on the time after fuel discharge from the reactor core. The dose rates above the SFP will not be affected by the expanded fuel pool storage capacity since the dose rate at the water surface is primarily due to the radionuclide concentration in the water, which is not expected to increase appreciably due to additional fuel storage. Therefore, the increased spent fuel storage capacity is expected to have a negligible effect on plant personnel exposures.

The total dose expended in performing the fuel pool reracking at LaSalle, Unit 2, was 11.1 person-rem. The licensee has estimated that the spent fuel reracking operation for Unit 1 will require the expenditure of approximately 8.55 person-rem, but has established a dose goal of 8.0 person-rem to complete the job. The following operations will contribute to the total dose of this job: (1) underwater disassembly of the existing racks (5.1 person-rem), (2) packaging of items removed from the pool (2.7 person-rem), and (3) installation of the new racks (0.75 person-rem). The licensee has incorporated several lessons learned from the Unit 2 rerack into the job planning for the Unit 1 rerack operation.

The Unit 1 SFP was pre-vacuumed in February 1992, in preparation for the reracking operation. The vacuum system currently used at LaSalle is more efficient than the pump and bag filter system which was used during the Unit 2 rerack. In order to increase the decontamination effectiveness, and reduce the number of components that will require hand wipedown after removal from the SFP, the existing Unit 1 spent fuel racks will be hydrolazed above the water surface of the pool. The racks will not need to be labeled (to facilitate possible reassembly) following removal from the pool, as was done during the Unit 2 rerack operation. This will reduce the number of person-rem needed for the packaging phase of the job. Also, unlike the racks installed in the Unit 2 SFP, the new racks will not require structural modifications prior to installation. This will result in an additional dose savings.

Prior to the removal of the existing racks from the SFP, all spent fuel assemblies in the Unit 1 pool will be transferred underwater to the Unit 2 SFP. Divers will then disassemble the existing Unit 1 racks into their components, which will then be cut into convenient lengths for removal from the SFP.

The licensee has incorporated several dose reduction techniques to ensure that the dose required to perform the Unit 1 SFP rerack will be significantly less than the dose required for the Unit 2 rerack. The licensee's goal of 8 person-rem for the Unit 1 rerack is consistent with the historical range of doses received during other SFP reracking operations, and is less than two percent of the average yearly dose for LaSalle (averaged over the years 1989-1991). On the basis of our review of the licensee's report, the staff concludes that the LaSalle Unit 1 SFP rerack can be performed in a manner that will ensure that exposure to workers will be as low as is reasonably achievable (ALARA) and is within the limits of 10 CFR Part 20.

2.6.2 Design Basis Accidents

The licensee evaluated the possible consequences of postulated accidents and included means for their avoidance in the design and operation of the facility, and has provided means for mitigation of their consequences should they occur. The licensee has evaluated the effect of the change on the calculated consequences of a spectrum of postulated design basis accidents and concluded that the effect of the proposed change is small and that the calculated consequences are within regulatory requirements and staff guideline dose values. The staff independently assessed these so-called design basis accidents (DBAs) and agrees with the licensee's conclusion that no previously unconsidered DBA would be created by the installation and use of the reracked SFP.

In NUREG-0519, the staff Safety Evaluation Report (SER) issued in March 1981 in support of the issuance of the LaSalle Unit 1 operating license, the staff conservatively estimated the offsite doses due to exposure to radionuclides released into the atmosphere as the result of a fuel handling accident. The fuel handling accident discussed in the SER is considered by the staff to be the bounding DBA for the spent fuel storage pool. The staff concluded in the SER that the mitigative features present at LaSalle would reduce DBA doses to well below the limits specified in 10 CFR Part 100.

Since the licensee, in their submittal, expressed interest in utilizing higher enrichment fuel, for which higher burnups are intended, the staff reevaluated the fuel handling accident for LaSalle to consider the effects of increased enrichments and burnups. In NUREG/CR-5009, issued in February 1988, the staff had previously found that increased fuel burnups could increase offsite doses from the fuel handling accident by a factor of 1.2. Fuel burnup to 60,000 megawatt-days/metric ton of uranium (MWD/MTU) would require the use of fuel initially enriched to approximately 5.0 w/o ^{235}U.

Therefore, in our reevaluation of the consequences of the fuel handling accident, the staff conservatively increased the previously estimated doses by a factor of 1.2. Table 2 compares the new and old doses resulting from a fuel handling accident with the dose limitations provided in 10 CFR Part 100 and the guidelines listed in NUREG-0800, Section 15.7.4.

The staff concludes that the only potential increased doses resulting from a design basis accident with extended fuel burnup of up to 60,000 MWD/MTU is the thyroid dose resulting from a fuel handling accident; these doses remain well within the exposure guideline of 300 rem to the thyroid set forth in 10 CFR Part 100. As discussed in Section 2.4 of this Safety Evaluation, the licensee has stated that it intends to use fuels with a maximum average enrichment of 4.25 w/o ^{235}U (4.6 w/o in the enriched fuel zone). Therefore, the staff concludes that the storage of fuels with increased enrichments of up to an average of 4.25 w/o ^{235}U (4.6 w/o in the enriched zone of the fuel) and fuel burnups of up to 60,000 MWD/MTU will not have an adverse radiological consequence, and are, therefore, acceptable.

3.0 STATE CONSULTATION

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

TABLE 2
Radiological Consequences of Fuel
Handling Design Basis Accident (rem)

	Exclusion Area		Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
Original Estimate (NUREG-0519)	<1	<1	<1	<1
Estimate for Higher Fuel Burnup*	1.2	<1	1.2	<1
Regulatory Requirement (NUREG-0800 Sect 15.7.4)	75	6	75	6

*Factor of 1.2 greater than original estimate for iodine

4.0 ENVIRONMENTAL CONSIDERATION

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 February 22, 1993 (58 FR 9576).

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

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6.0 REFERENCES

(1) Letter from CECo, "LaSalle County Station, Unit 1, Application for Amendment to Facility Operating License NPF-11, Attachment A, Technical Specifications, Unit 1 Spent Fuel Pool Rerack," dated June 5, 1992.

(2) Letter from CECo transmitting Figure 4.6.1, "Reactivities in the Storage Rack with 4.4% and 4.6% Enriched Fuel," dated July 7, 1992.

(3) Letter from CECo transmitting responses to questions raised during July 13, 1992, telecon, dated July 20, 1992.

(4) Letter from CECo transmitting responses to NRC's September 30, 1992, request for additional information, dated November 4, 1992.

(5) American Concrete Institute (ACI) 318-71, "Building Code Requirements for Reinforced Concrete."