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Docket Nos. 50-254
50-265

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

The Commission has issued the enclosed Amendment Nos. 79 and 73 to Licenses Nos. DPR-29 and DPR-30 for Quad Cities Station Units Nos. 1 and 2. These amendments consist of changes to the Technical Specifications and are in response to your letter dated March 26, 1981, supplemented by letters dated June 24, July 24, August 10, August 26, October 19, November 2 and December 8, 1981, January 27 and March 12, 1982.

These amendments allow an increase in the spent fuel storage capacity at the Station from 2920 to a maximum of 7684 assemblies by use of neutron absorbing spent fuel storage racks.

Although the Safety Evaluation and Environmental Impact Appraisal supporting this Amendment were sent to you when they were issued April 9, 1982, copies of these supporting documents are enclosed, together with the Notice of Issuance and Negative Declaration for this action. Please note that page 1 of the Safety Evaluation and page 4 of the Environmental Impact Appraisal have been changed to agree with the correct submittal dates indicated above.

Sincerely,

Original signed by

Roby Bevan, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosure:
1. Amendment No. 79 to DPR-29
2. Amendment No. 73 to DPR-30
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice

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See next page

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DATE	5/28/82	6/1/32	6/2/82	6/3/82	6/7/82		

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 79
License No. DPR-29

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 26, 1981 as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 79, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 9, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Revise the Appendix "A" Technical Specifications by removing page 5.0-1 and replacing with the attached revised page 5.0-1.

5.0 DESIGN FEATURES

5.1 Site

The Quad-Cities Station, which consists of a tract of land of approximately 404 acres, is located about 3 miles north of Cordova, Illinois, Rock Island County, Illinois. The tract is situated in portions of Sections 7, 8, 17, and 18 of Township 20 North, Range 2 East.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies.

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide power (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR, and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 24% of gravity. Dynamic analysis was used to determine the earthquake acceleration application to the various elevations in the reactor building.

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. DPR-30

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated March 26, 1981 as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 73 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 9, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 73

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix "A" Technical Specifications by removing page 5.0-1 and replacing with the attached revised page 5.0-1.

5.0 DESIGN FEATURES

5.1 Site

The Quad-Cities Station, which consists of a tract of land of approximately 404 acres, is located about 3 miles north of Cordova, Illinois, Rock Island County, Illinois. The tract is situated in portions of Sections 7, 8, 17, and 18 of Township 20 North, Range 2 East.

5.2 Reactor

A. The core shall consist of not more than 724 fuel assemblies.

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide power (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR, and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.95.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 24% of gravity. Dynamic analysis was used to determine the earthquake acceleration application to the various elevations in the reactor building.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. DPR-29 AND

FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT NOS. 1 AND 2

DOCKET NOS. 50-254 AND 50-265

Authors: R. Bevan; S. Block; J. Boegli; W. Brooks; F. Clemenson; O. Rothberg; B. Turovlin; and P. Wu

1.0 INTRODUCTION

By letter dated March 26, 1981, and supplemented by letters dated June 24, July 24, August 10, August 26, October 19, November 2, and December 8, 1981, January 27 and March 12, 1982, Commonwealth Edison Company (CECo, the licensee) requested amendments to Facility Operating Licenses DPR-29 and DPR-30 for Quad Cities Station, Units 1 and 2, respectively. The request is to authorize increased storage capability in the spent fuel pools (SPF) for the two nuclear units. The proposed modifications would increase the SFP storage spaces from the currently licensed 2920 spaces to 7684 spaces combined total for the two pools. This expanded storage capacity will allow the continued operation of the two nuclear units with onsite storage of spent fuel to past the year 2000. The licensees basic supporting document for this action is a report, Spent Fuel Pool Modification for Increased Storage Capacity, Quad Cities Nuclear Unit 1, Docket No. 50-254, and Quad Cities Nuclear Unit No. 2, Docket No. 50-265, Rev. 1, dated June, 1981.

2.0 DISCUSSION

The licensee's proposal would increase the SFP storage capacity by replacing the existing spent fuel storage racks with new high density storage racks. The new racks will contain neutron absorber material in the rack walls so that spacing between stored assemblies can be reduced while maintaining adequate criticality margin.

The high density racks are made up of modules, each module being composed of six-inch square cells, each cell accommodating a single BWR fuel assembly. The cell walls contain a neutron absorber material sandwiched between sheets of stainless steel. The cells making up the module have 6.22-inch center-to-center spacing. The general arrangement of the modules in the pools is shown in Figures 2.1 and 2.2 of the licensee's application and basic supporting document. The general details of

design and construction of the racks are contained in Figures 3.1 through 3.8 and are described in Section 3 of the licensees basic supporting document. The racks are free standing in that they are neither anchored to the floor of the pool or walls, nor are the modules interconnected.

The applicable codes, standards, and practices for this modification are set forth in Section 3.2 of the licensee's basic supporting document. A detailed structural analysis is described in Section 6 of the document to show the adequacy of the racks to resist the postulated stress combinations for normal and postulated accident conditions. Section 9 of the licensee's basic supporting document describes the detailed analysis to show that the pool floor meets all structural acceptance requirements when conservatively analyzed.

The safety considerations associated with this proposed action are addressed below. A separate environmental impact appraisal has been prepared for this action.

3.0 EVALUATION

3.1 Structural and Mechanical Design Considerations

Description

Quad Cities Units 1 and 2 each have fuel storage pools 33 feet wide x 41 feet long. The Unit 1 pool will contain 19 high density fuel racks in seven different module sizes with a total of 3714 storage locations, while the Unit 2 pool will contain 3970 storage cells arranged in 20 racks with six different module sizes in this pool.

All modules are free standing, i.e., they are not anchored to the pool walls. The minimum gap between adjacent racks is three inches at all locations and nine inches between the racks and the fuel pool walls. Because of these gaps, the possibility of inter-rack impact, or rack collision with pool wall hardware during the postulated ground seismic motion, is precluded.

The racks will be constructed from ASTM 240 - 304, austenitic steel sheet material, ASTM 204-304 austenitic steel plate material, and ASTM 182 - F304 austenitic steel forging material. A typical module contains storage cells which have 6 inch minimum internal cross-sectional opening. Skip welding at the top ensures proper venting of the sandwiched space in the sub-elements which make up the fuel racks.

The rack assembly is typically supported on four plate-type supports. The supports elevate the module base plate 6.5 inches above the pool floor level, thus creating the water plenum for coolant flow.

Further details of the spent fuel racks are illustrated in the licensee's basic supporting document.

Evaluation and Conclusions

In our evaluation of the licensee's proposed action, established codes, standards and criteria were applied, consistent with the NRC's guidance, "OT Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Application," dated April, 1978 and revised January, 1979. Accordingly, the design of the racks, fabrication, and installation criteria; the structural design and analysis procedures for all loadings, including seismic and impact loadings; the load combinations; the structural acceptance criteria; the quality assurance requirements for design, and applicable industry codes were all reviewed in accordance with the applicable portions of that NRC guidance.

For the design of the spent fuel modules, two sets of criteria were to be satisfied. The first establishes requirements to ensure that adjacent racks will not impact during the Safe Shutdown Earthquake (SSE), assuming the lower bound value of the pool surface friction coefficient. It is required by this criterion that the factors of safety against tilting be 1.5 for the OBE and 1.1 for the SSE. The second set of criteria establishes requirements to ensure that loading combinations and stress allowables are in accordance with Section III. Subsection NF of the ASME 1980 Edition. The basic material allowables, fabrications, installations and quality control of the modules also conform with the same code. The loading considered in the analysis involves dead loads, live loads, thermal loading, and seismic loadings (OBE or SSE). Additional analyses were performed to evaluate the effects of a postulated accident involving the dropping of a fuel assembly on the racks and on the fuel pool liner, and the fuel handling crane uplift accident.

A dynamic analytical model, consisting of beams, gaps, springs, dampers and inertia coupling representing fluid coupling between rack and assemblies, and between rack and adjacent racks, was used to predict the maximum sliding distance and seismic forces resulting from the SSE. These forces were then used to predict the seismic stresses and displacements. The coefficient of friction between the stainless steel liner and the leveling legs of the racks used in the analysis was chosen based on the information contained in a report by E. Rabinowicz of Massachusetts Institute of Technology entitled "Friction Coefficients of Water Lubrication Stainless Steel for a Spent Fuel Rack Facility" dated November 5, 1976. The result of this analysis indicates that, although the proposed racks which are free-standing may slide toward each other during the SSE, sufficient gaps are provided between the modules and the modules and the pool walls such that the inter-rack impact, or the rack collision with the pool walls, is precluded.

The analysis, design, fabrication, and criteria for establishing installation procedures of the proposed new spent fuel racks are in conformance with accepted codes, standards and criteria identified in the NRC guidance. The structural design and analysis procedures for all loadings, including seismic, thermal, and impact loading; the acceptance criteria for the appropriate loading conditions and combinations; and the applicable industry codes are in accordance with appropriate sections of the NRC staff "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

Allowable stress limits for the combined loading conditions are in accordance with the ASME Code, App. XVII. Yield stress values at the appropriate temperature were obtained from Section III of the ASME Code. The quality assurance and criteria for the materials, fabrication and installation of the new racks are in accordance with accepted requirements of the ASME Code.

The effects of the additional loads on the existing pool structure due to the new fuel racks, existing fuel racks, and equipment have been examined. The pool structural integrity is assured by conformance with the Standard Review Plan Section 3.8.4.

Results of the seismic and structural analyses indicate that the racks are capable of withstanding the loads associated with all design loading conditions. Also, impact due to fuel assembly/cell interaction has been considered, and will result in no damage to the racks or fuel assemblies.

Two types of postulated fuel assembly drops onto the racks were analyzed by the licensee and evaluated by the staff. The first drop is a straight drop of a fuel assembly from a maximum of 36 inches above the storage location and impacting the base. The second drop involves a fuel assembly dropping from a maximum of 36 inches above the rack and hitting the top of the rack. In both cases, the impact energy is dissipated by local yielding; however, the sub-criticality of the fuel arrays is not violated.

The dropping of a heavy load onto the protective pool liner of the pool floor was also analyzed. Although local damage and plastic deformation may occur, the overall structural integrity of the liner is maintained.

The effect of postulated stuck fuel assembly due to the attempted withdrawal was considered, and the damage, if any, was required to be limited to the region above the active fuel elements. Results of the stuck fuel assembly analysis show that the stress is below that allowed for the applicable loading combinations.

We find that with respect to structural and mechanical design the subject modification proposed by the licensee satisfies the applicable requirements of General Design Criteria 2, 4, 61, and 62 of 10 CFR, Part 50, Appendix A and is acceptable.

3.2 Materials Considerations

Discussion and Evaluation

We have reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water. In addition, our review has included an evaluation of the Boraflex neutron absorber material used in the high density storage locations for environmental stability.

There will be both the old and the new types of spent fuel storage cells in the Quad Cities Station spent fuel pools during the transition time while new storage modules are being installed. The transition period is expected to last slightly over one year. The spent fuel pool is filled with demineralized high-purity, high resistivity water.

The new high-density spent fuel storage racks are of welded stainless steel construction with a "Boraflex" neutron absorber sandwiched between the stainless steel sheets. The neutron absorber is composed of boron carbide powder in a rubber-like silicone polymeric matrix.

The old low density fuel storage tubes provide for the interim storage of fuel assemblies and are constructed of aluminum without neutron absorber material. The anticipated corrosion of the aluminum alloys, type 1100 or 6061, is negligible in water of spent fuel pool quality at temperatures up to the boiling point of water; at 125 C (257 F) a corrosion rate of 1.5×10^{-4} mils/day has been measured for alloy 6061 aluminum, in water of pH 7, which corresponds to a total corrosion of 1.1 mils in twenty years. Since the oxidation rate will continue to decrease slightly over this period, this estimate is considered to be conservative.

The inherent high corrosion resistance of aluminum and stainless steel makes them well suited for use in demineralized water. Aluminum and stainless steel fuel storage racks submerged in water have been in use for 10 years with no deterioration evident.

Aluminum and 300-series stainless steel are very similar insofar as their coupled potential is concerned. Because the pool water has very low conductivity, galvanic corrosion should not occur. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry.

The pool liner, rack lattice structure and the high density fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated high purity water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.0×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar galvanic potentials. The Boraflex poison material is composed of non-conductive materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material.

The space which contains the Boraflex is vented to the pool. Venting will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the stainless steel tube.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell surveillance program. Surveillance samples are in the form of removable stainless steel clad Boraflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically.

Conclusions

From our evaluation as discussed above we conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the remaining life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

We further conclude that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate, based on test data and actual service experience in operating reactors.

We have reviewed the surveillance program and we conclude that the monitoring of the materials in the spent fuel storage pool, as proposed by the licensee, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. We therefore find that the implementation of a monitoring program and the selection of appropriate materials of construction by the licensee meet the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, by preventing criticality by maintaining structural integrity of components and of the boron poison.

3.3 Installation and Heavy Load Handling Considerations

The results of the staff's generic review of handling heavy loads at nuclear power plants, i.e., NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," is ongoing and will not be completed before the spent fuel pool modifications are to commence. Therefore, we have limited this review and evaluation to the heavy load handling operations associated with the Quad Cities Unit 1 and 2 proposed spent fuel modifications.

The heaviest identified load with this modification is a 16 x 16 storage rack weighing 16 1/2 tons, whereas the main hoist on the reactor building crane is rated at 125 tons. The overhead crane was previously modified and as documented in a NRC review dated January 27, 1977, we found it to be acceptable. From this we conclude that the overhead load handling system is acceptable.

The licensee has stated that the travel paths of the storage racks will be established before moving the racks, and the travel paths will be based on the studies associated with NUREG-0612. The handling procedures will be such that none of the storage racks containing stored fuel will be immediately adjacent to the empty rack being moved. Consequently, a load handling mishap will not impact on stored fuel. Based on these considerations, we conclude the procedures are acceptable.

The June 22, 1981 Commonwealth Edison response to our December 22, 1980 generic letter on control of heavy loads states that operator training qualifications and conduct for Quad Cities Units 1 and 2 comply with ANSI B30. 2-1976. From this we conclude the qualifications and conduct of operators handling heavy loads are acceptable. The above submittal also states that the inspection, testing and maintenance related to Quad Cities cranes comply with ANSI B30.2-1976. From this we conclude that adequate measures will be taken to assure the operability of the cranes used in handling the spent fuel pool modifications loads, and are therefore, in this respect acceptable.

A lifting yoke has been designed to handle the new storage racks. It will consist of a four-leg bridle hitch with turnbuckles, attached to a rectangular frame that supports four lifting rods that will be threaded into the four legs of the racks. The holes in the rectangular frame permit the lifting rod spacing to be adjusted so as to permit them to remain vertical and yet accommodate the seven different sized racks. Figure 3-8 of the licensee's submittal indicates the lifting yoke is rated for 22.7 tons while the heaviest storage rack is 16 1/2 tons. Based on the above, we conclude that the lifting yoke is adequate for handling the new storage racks, and therefore, acceptable.

The existing aluminum open lattice storage racks will be removed using the overhead crane and a wire rope sling. The sling design complies with the requirements of ANSI B30.9-1971. It's load rating is slightly more than twice the weight of the heaviest rack to be removed. The ends of the sling terminate with locking safety hooks which are attached to lifting lugs on the storage rack. Based on the above we conclude that rigging interposed between the crane hook and the load is acceptable for handling the old storage racks, and that the crane meets the objectives of APCSB BTP 9-1 and has sufficient capacity for the described operations. The travel paths, procedures, operator training and crane maintenance are adequate to accomplish the heavy load handling operations associated with spent fuel pool modifications and are therefore acceptable.

In regard to the handling of light loads over stored spent fuel, an analysis has been made assuming the channel measuring device, weighing 1000 pounds, was dropped 30 feet above the racks. The results indicate that deformation will occur but the k_{eff} remains equal to or less than 0.95, in conformance with SRP, Section 9.1.2. In this respect we find that a postulated light load drop will not cause a criticality accident.

The proposed modifications meet the guidelines of the applicable portions of the following: Regulatory Guides 1.13, 1.29 and 1.71, 1.85, 1.92 and 1.124; and 10 CFR Part 50, Appendix A, General Design Criteria 1, 2, 61, 62 and 63; Standard Review Plan Sections 3.8.3 and 3.8.4 and industry standards ANSI N210-1976, ACI 318-77, AISC, ASTM, ASME Section III Division I Subsection NF 1980 and ASME Section IX-1980.

3.4 Criticality Considerations

Discussion and Evaluation

The boron content in the neutron absorber material in the rack walls is equivalent to a B-10 areal density of 0.01728 grams per square centimeter. The multiplication factor of the racks is calculated for an 8 x 8 assembly having a uniform enrichment of 3.2 weight percent U-235. The infinite multiplication factor for this assembly in the standard reactor configuration at cold clean conditions is 1.362. For comparison the maximum value of the infinite multiplication factor for reload bundles is 1.241 at the most reactive point in the bundle life (NEDO-24011-P-A,"General Electric Generic Reload Fuel Application" Amendment 9, dated November 17, 1980).

The rack design is thus conservative for assemblies which are anticipated to be stored in the racks. Other conservatisms present in the analysis include the use of the minimum (worst case) center-to-center spacing and a Boraflex poison plate width less than the design value.

The criticality analyses of the racks were performed with the AMPX-KENO computer code package using the 123 group XSDRN cross-section set with the NITAWL subroutine for U-238 resonance shielding effects. This code has been benchmarked against experiments by Southern Science Applications, Inc. and the results are reported in SSA-127 (Rev. 1), "Benchmark Calculations for Spent Fuel Storage Racks" dated September 1980. The results of the comparison show that the Code set underpredicts the multiplication factor by 0.36 percent reactivity change with a deviation of 1.23 percent reactivity change at the 95 percent probability, 95 percent confidence level. Trend analyses were performed to obtain an estimate of the effect of varying amounts of boron between assemblies. This analysis showed that AMPX-KENO should overpredict the reactivity of the Quad Cities racks by 3.1 $\pm$ 1.2 percent reactivity change. No credit is taken for this overprediction in the analysis.

Sensitivity analyses were performed to obtain the reactivity effect of the variation of stainless steel wall thickness, boron loading variations, and channel deformation (bulge). The results of these studies indicate a total uncertainty of 0.97 percent reactivity change due to these effects.

The calculated value of the nominal case multiplication factor was 0.9155 ± .0067 where the uncertainty is the statistical uncertainty in the Monte-Carlo (KENO) calculation only. To this value must be added the calculational bias of 0.0036 and the statistical combination of the bias uncertainty (0.0123), the calculational uncertainty (0.0067) and the mechanical uncertainty (0.0097). The resulting value for the maximum multiplication factor is 0.9361. This value meets the acceptance criterion that requires the k_{eff} be less than or equal to 0.95.

The criticality effects of various abnormal and postulated accident conditions have been investigated. This includes improper positioning of an assembly in its storage rack, bowing of the channel, variations in pool temperature, a dropped fuel assembly, and a missing absorber plate in the racks. These analyses show that the criticality acceptance criterion is not violated when not more than one Boraflex plate out of fifteen is missing. Appropriate measures will be taken during manufacture of the racks and prior to installation in the pool to assure the presence of the boron absorber material as designed.

In the course of our review, we have found that:

1. State-of-the-art calculation methods which have been benchmarked against critical experiments have been used,

2. Credible abnormal configurations have been investigated,

3. Uncertainties and biases have been treated, and

4. The result, including all uncertainties, meets our acceptance criteria for the nominal case and for abnormal and postulated accident conditions.

From the above considerations, we find that fuel assemblies of the 8 x 8 two-water rod design, having average enrichments less than or equal to 3.2 weight percent U-235, other fuel designs containing less than 15.49 grams of U-235 per axial centimeter, or BWR assemblies having cold clean infinite multiplication factors in the Quad Cities reactor geometry of less than 1.36 may be safely stored in the Quad Cities 1 and 2 storage pool.

Conclusion

We conclude that any number of spent fuel assemblies of a design likely to be used in the Quad Cities reactors can be safely stored in the spent fuel racks with adequate criticality margin.

3.5 Spent Fuel Pool Cooling Considerations

Description and Evaluation

Quad Cities Units 1 and 2 each has a stainless steel lined reinforced concrete spent fuel storage pool. The two pools are joined by a transfer canal. Fuel can be transferred between the two pools via the transfer canal after opening the two gates, located at the sides of the respective pools. A normal fuel discharge, i.e., about 200 assemblies, occurs at 18 month intervals. To the extent possible the discharge cycles of the two units are phased such that the refueling operations on the two units will not occur simultaneously.

Separate spent fuel pool cooling systems are provided for each of the two pools. The FSAR states that each of the two separate cooling systems was designed to be capable of maintaining the pool water temperature of their respective pools below 125 degrees F during maximum normal discharges, when the reactor building closed cooling water system is at its maximum temperature of 105 degrees F. This assures that a comfortable working environment can be maintained during normal conditions. Further, on those infrequent off normal conditions where, for example, a full core discharge occurs, the pool water temperature will not exceed 150 degrees F. Analyses of the pool water temperatures following this proposed spent fuel expansion shows the maximum pool water temperatures does not exceed 134.6 degrees F when the pool is completely filled with normal discharges. This is nearly a 10 degree increase over that stated in the FSAR. This is less than the 140 degrees F limit given in the Standard Review Plan Section 9.1.3 - Spent Fuel Pool Cooling and Cleanup System and is acceptable. Further, the analysis of the maximum pool water temperature following a full core discharge, at any point until the pool is filled with spent fuel, will not exceed 145.4 degrees F. This is less than the 150 degrees F stated in the FSAR, and is acceptable.

The spent fuel pool cooling system (SFPCS) for each unit consists of one cooling loop having two parallel, 50 percent capacity, pumps placed in series with two, 50 percent capacity, parallel heat exchangers. Each pump is rated at 700 gpm, i.e., 350,000 pounds per hour, and assuming the pool water temperature is at 125 degrees F each heat exchanger is rated at 3.65×10^6 BTU/hr. Therefore each unit's spent fuel pool cooling system has a total design flow of 700,000 pounds per hour and a total heat removal capability of 7.3×10^6 BTU/hr at a pool water temperature of 125 degrees F. By allowing the pool water temperature to rise to 134.6 degrees F the total heat removal capability of each spent fuel pool cooling system increases to approximately 10.9×10^6 BTU/hr.

In addition to the above spent fuel pool cooling system, provisions have been made to cross tie the spent fuel pool cooling system to the residual heat removal (RHR) system. This is accomplished by installing two 6 inch pipe size spool pieces in the two legs of the spent fuel pool cooling loop. The six inch RHR tie-in line will provide an additional spent fuel pool cooling water flow of 1,000 gpm i.e., 500,000 pounds per hour. While it has not been stated by the licensee, we note that it appears feasible to use the cooling system in one unit to assist cooling the pool water in the adjacent unit pool. This could be accomplished by opening the two gates in the transfer canal and allowing an interchange of water between the two pools.

Decay Heat

The licensee has analyzed five different cases of spent fuel pool decay heat loads and the resultant pool water temperatures with and without the additional cooling provided by the residual heat removal system (RHR).

The cases investigated are as follows:

(1) The pool is filled with normal discharges of 240 fuel assemblies and cooling is only provided by the SFPCS (decay heat equals 11.2×10^6 BTU/hr).

(2) The pool is filled with normal discharges of 240 fuel assemblies and cooling is provided by the SFPCS and the RHR system (decay heat equals 11.2×10^6 BTU/hr).

(3) The pool is filled with normal discharges of 200 fuel assemblies and cooling is provided only by the SFPCS (decay heat equals 9.65×10^6 BTU/hr).

(4) The pool is filled with normal discharges of 200 fuel assemblies and cooling is provided by the SFPCS and the RHR system (decay heat equals 9.65×10^6 BTU/hr).

(5) The pool is filled, with normal discharges plus a recently discharged full core and cooling is provided by the SFPCS and RHR system (decay heat equals 24.7×10^6 BTU/hr).

In the case of normal discharges and a full core discharge it is assumed 100 hours will be required to prepare the reactor for refueling. The transfer of a normal discharge of either 200 or 240 assemblies can be accomplished in two days. In the case of a full core discharge, six days will be required to transfer the fuel to the storage pool.

According to the licensees analysis, the maximum bulk temperature of the pool will not exceed 134.6 degrees when a normal fuel discharge of spent fuel is placed in the pool. Although no safety problem is created by a somewhat higher pool temperature, the higher temperature encroaches upon margin assumed in our analysis of the licensee's ability to provide makeup water in the event that pool cooling capability is lost. Similarly, in the event of a full core discharge to the pool, the licensees analysis shows that the pool temperature will not exceed 145.4 degrees. Should the pool bulk temperature exceed this value during a full core discharge, further placement of spent fuel into the pool should be suspended until the temperature is brought to below 145 degrees F. The licensee has agreed to include this limit in its operating procedures.

Makeup Water

The spent fuel pool system is designed to minimize the loss of water from the pool and to prevent the water level from falling below a safe level above the stored fuel. For example all penetrations into the pool, except for valved drains, are located at a height such that there will always be a safe level of water above the fuel. Each pool has a high and low water level monitor. Both monitors actuate local annunciators and the low level monitor also actuates a control room low level annunciator. In the event makeup water is needed, there are two sources of makeup water, the condensate storage tanks and the fire system. Approximately 550 gpm of condensate water can be delivered to the pools via the condensate transfer pumps and skimmer surge tanks within a few minutes. In addition as much as 1,000 gpm of condensate storage tank water can be supplied to the pools using the RHR pumps following the installation of a spool piece joining the RHR system to the spent fuel pool cooling system. About three hours would be required to install the spool piece.

In the event that the above identified sources of water become unavailable, the fire system hoses are capable of providing makeup water from the river within approximately 30 minutes. The two pumps, each rated at 3,200 gpm, can provide water to the pool far in excess of any reasonable need.

We conclude the makeup water system is adequate and acceptable because makeup water is available from the condensate storage tanks and river via the fire system, and their respective makeup rates exceed the boil off rate described below. Further, this makeup water can be made available before boiling would occur.

Boil Off Rate

The minimum time before boiling occurs and the maximum boil off rate were established assuming that: (1) the heatup follows a full core discharge in Unit 2 storage pool (i.e., the pool with the least water inventory of 44, 471 ft^3 of water), (2) the pool water bulk temperature is at its maximum temperature of 145.4 degrees F, (3) there is no exchange of water between Pool 1 and Pool 2, (4) all pool cooling is lost and (5) no credit is taken for heat lost to the pool walls and floor. Under the above conditions about 7 1/2 hours would elapse before bulk boiling would occur. The maximum boiloff rate would be 51 gpm.

Based on the above, we conclude that the available sources of makeup water are adequate, the time required to activate the makeup system is sufficiently less than the time required to reach boiling, and the makeup rates from both makeup sources exceed the boil off rate, and therefore the provisions for makeup water are acceptable.

Local Boiling

Using a conservative thermal hydraulic circulation model of pool water flowing down along the walls, laterally across the pool floor in the water plenum and up through the stored fuel assemblies, the maximum calculated water temperature at the outlet of the fuel assemblies was shown not to exceed 167 degrees Fahrenheit.

The saturation temperature at this point is 240 degrees F. Due to the margin between these two temperatures we conclude that nucleate boiling will not occur and in this respect the design is acceptable.

Conclusion

Cooling capability for the spent fuel pools for the two nuclear units has been evaluated for the maximum expected loading conditions for the new racks. We conclude that the presently installed pool cooling capability is adequate to handle the heat load under any reasonably expected conditions of operation.

3.6 Spent Fuel Pool Cleanup System

Description and Evaluation

The spent fuel pool cleanup system consists of a filter demineralizer (precoat filter material and powdered anion and cation resin), filters, and associated piping, valves, and fittings. The system is designed to remove corrosion products, fission products, and impurities from the pool water. Pool water purity is monitored by a continuous conductivity meter installed on the inlet to the fuel pool demineralizers, and by periodic grab samples for laboratory analysis. Once a week a representative grab sample is obtained from the fuel pool demineralizer inlet line for pH, for chloride, silica, and turbidity analysis. Weekly activity checks are made for gross beta and gross alpha activity. Once a month a sample from the same location is obtained for a gamma isotopic analysis. All peaks are identified. All identified isotopes are quantified, and an LLD is determined for Kr-85.

The criterion for a demineralizer backwash and precoat is a consistent excursion from the chemistry limits, or high differential pressure [25 psid] across the demineralizer. We agree with the licensee that the proposed high density fuel storage will not significantly alter the chemistry or radiochemistry of the spent fuel pool water.

Past experience shows that the greatest increase in radioactivity and impurities in spent fuel pool water occurs during refueling and spent fuel handling. The refueling frequency, the amount of core to be replaced for each fuel cycle, and frequency of operating the spent fuel pool cleanup system are not expected to increase as a result of high density fuel storage. The chemical and radionuclide composition of the spent fuel pool water is not expected to change as a result of the proposed high density fuel storage. Past experience also shows that no significant leakage of fission products from spent fuel stored in pools occurs after the fuel has cooled for several months. To maintain water quality, the licensee has established the frequency of chemical and radionuclide analysis that will be performed to monitor the water quality and the need for spent fuel pool cleanup system demineralizer resin and filter replacement. In addition, the licensee has also set the chemical and radiochemical limits to be used in monitoring the spent fuel pool water quality and initiating corrective action. We agree with the licensee that the increased quantity of spent fuel to be stored will not contribute significantly to the amount of radioactivity from fission products in the spent fuel pool water.

The proposed expansion of the spent fuel pool will not appreciably affect the capability and capacity of the existing spent fuel pool cleanup system. More frequent replacements of filters or demineralizer resin, required when the differential pressure exceeds 25 psid or decontamination effectiveness is reduced,as indicated by the licensee, can offset any potential increase in radioactivity and impurities in the pool water as a result of the expansion of stored spent fuel. Thus we have determined that the existing fuel pool cleanup system with the proposed high density fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool and thus meets the requirements of General Design Criterion 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate fuel storage systems, (2) is capable of reducing occupational exposures to radiation by removing radioactive products from the pool water, and thus meet the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as reasonably achievable; (3) confines radioactive materials in the pool water into the filters and demineralizers, and thus meets Regulatory Position C.2.f(c) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from pool water by filters, and thus meets Regulatory Position C.2f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action.

Conclusion

On the basis of the above evaluation, we conclude that:

(1) The existing spent fuel pool cleanup system meets General Design Criterion 61 of 10 CFR Part 50, Appendix A, Section 20.1(c) of 10 CFR Part 20 and the appropriate Sections of Regulatory Guide 8.8 and, therefore, is acceptable for the proposed high density fuel storage.

(2) The existing spent fuel pool cleanup system is adequate for the proposed modification.

(3) The conclusions of the evaluation of the waste treatment systems as found in the NRC staff's Quad Cities, Unit Nos. 1 and 2, Safety Evaluation Report (August 25, 1971), are unchanged by the modification of the spent fuel storage system.

3.7 Occupational Radiation Exposure

Description and Evaluation

We have reviewed the licensee's plan for the removal and disposal of the low density racks, and installation of the high density racks, with respect to occupational radiation exposure. The occupational exposure for this operation is estimated by the licensee to range from

18 to 39 man-rem. This estimate is based on the licensee's detailed breakdown of occupational exposure for each phase of the modification. The licensee considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job is being performed. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. One potential source of radiation is radioactive activation or corrosion products called crud. Crud may be released to the pool water because of fuel movements during the proposed modification. This could increase radiation levels in the vicinity of the pool. During refuelings, when the spent fuel is first moved into the fuel pool, the addition of crud to the pool water from the fuel assembly and from the introduction of primary coolant to the pool water is greatest. However, the licensee does not expect to have significant releases of crud to the pool water during modification of the pools. The purification system for the pool, which has kept radiation levels in the vicinity of the pool to low levels, includes a filter to remove crud and will be operating during the modification of the pool.

The licensee has presented three alternative plans for removal and disposal of the old racks. These are (1) to crate and ship intact racks to a licensed burial facility; (2) to cut the racks into small pieces with a shredder and pack the pieces into drums for burial at a licensed burial facility; and (3) to have an outside vendor chemically decontaminate the intact racks. If the decontamination option is selected, the decontamination chemicals would be reduced in volume, solidified and buried. The bulk of the decontaminated racks could be disposed of as clean scrap. This last alternative is to be tested at the Dresden station and results of that work will be influential in the final decision. In any event, the disposal methodology will follow "as low as reasonably achievable" (ALARA) guidelines for each of the alternatives. It should be noted that the procedures for removal of old racks from the pool will be performed independent of the aforementioned disposal alternatives. The racks will be individually lifted from the pool water and rinsed by hydrolasing to remove any loose radioactivity that will drip back into the pool water prior to movement to a receiving area for preparation for disposal.

Divers will be used for setting and shimming the high density racks. Related experience from the Dresden SFP modification indicates that the diver exposure should be less than 2 man-rem for rack installation including clean-up and diver work.

Conclusion

Based on our review of the manner in which the licensee will perform their modification, and related experience from other operating reactors that have performed similar spent fuel pool modifications, we conclude that the Quad City spent fuel pool modification can be performed in a manner that will ensure as low as is reasonably achievable (ALARA) exposures to workers.

4.0 CONCLUSION

We have performed an evaluation of the licensee's proposed modifications based primarily on information provided to us in the licensee's basic supporting document. This document has been revised and supplemented during the course of our review in response to staff questions, and from meetings and discussions with the licensee, and to address new or more refined information regarding the proposed modification.

Our evaluation concludes that the proposed modification of the Quad Cities Station Units 1 and 2 spent fuel storage is acceptable because:

(1) The structural design and the materials of construction are acceptable.

(2) The installation and use of the proposed fuel handling racks can be accomplished safely.

(3) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool is sufficiently small that no additional restrictions on load movement are necessary while our generic review of the issues is underway.

(4) The installation and use of the new fuel racks does not alter the potential consequences of the design basis accident for the SFP, i.e., the rupture of all the fuel pins in the equivalent of a single fuel assembly and the subsequent release of the radioactive inventory within the gap of each fuel pin, as already reviewed and approved in the FSAR for Quad Cities Station.

(5) The physical design of the new storage racks will preclude criticality for any credible moderating condition.

(6) The cooling system for each of the spent fuel pools has acceptable cooling capacity.

(7) The conclusions of the evaluation of the waste treatment systems are unchanged by the modification of the spent fuel pool.

(8) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the spent fuel pool would be negligible.

We conclude, then, 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the proposed license amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 9, 1982

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

ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO THE MODIFICATION OF THE SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS

AND ELECTRIC COMPANY

QUAD CITIES STATION UNIT NOS. 1 AND 2

DOCKET NOS. 50-254 AND 50-265

APRIL 9, 1982

1.0 Introduction and Discussion

The combined spent fuel storage capacity of the two nuclear units at Quad Cities Station was originally 2280 fuel assemblies, or storage for 1 3/5 cores from each of the two units. This licensed capability was later increased to 2920 assemblies, although little or no actual increase in installed storage capacity was made. This limited storage capability was in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a period of 3 to 5 years and then shipped offsite for reprocessing and recycling of the fuel.

Reprocessing of spent fuel did not develop as had been anticipated, however, and in September, 1975, the Nuclear Regulatory Commission (NRC, the Commission) directed the NRC staff (the staff) to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement would consider alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August, 1979. In the FGEIS, consistent

with the long range policy, the storage of spent fuel is considered to be interim storage, to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of onsite fuel storage capacity by modification of the existing spent fuel pools. Applications for fifty such spent fuel capacity increases have been reviewed and approved. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing spent fuel pools, other spent fuel storage alternatives are discussed in detail in the FGEIS. The finding of the FGEIS is that the environmental impact costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of the various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, that of shutting down the reactor when the spent fuel storage capacity is filled, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) addresses the environmental concerns related only to expansion of the Quad Cities Station spent fuel storage pools. Additional discussion of the alternatives to increasing the storage capacity of existing spent fuel pools is contained in the FGEIS.

1.1 Description of the Proposed Action

By application dated March 26, 1981, and supplemented by letters dated June 24, July 24, August 10, August 26, October 19, November 2, December 8, 1981, January 27 and March 12, 1982, Commonwealth Edison proposed an amendment that would allow an increase in the licensed storage capacity of the two spent fuel pools from 2,920 to 7,570 fuel assemblies. The storage capability would be increased by replacing the existing racks with new, more compact, neutron absorbing racks. This would provide storage for spent fuel generated at Quad Cities for the next 20 years.

The environmental impacts of Quad Cities Station, as designed, were considered in the NRC's Final Environmental Statement (FES) issued September, 1972, relative to the continuation of construction and operation of the Station. The licensee was later authorized to increase the storage capacity from 2280 to 2920 bundles. The environmental impact of this action was considered in an environmental impact appraisal issued with our authorization

for this action in January, 1978.

In this EIA we have evaluated any additional environmental impacts which are attributable to the currently proposed increase in the SFP storage capacity for the Station.

1.2 Need for Increased Storage Capacity

Spent fuel storage pools are provided for each of the two nuclear generating units at the Quad Cities Station. The Station now has a combined licensed fuel storage capacity of 2920 spaces. Of this number, 2280 spaces are provided by racks already installed. Of the installed racks, 1716 spaces are occupied by spent fuel and 564 spaces are empty. For the Unit 1 refuel outage now scheduled for fall, 1982, the full core of 724 assemblies needs to be removed and stored temporarily in order to safely and with minimum personnel exposure perform needed inspections and modifications. The 564 empty spaces in the racks now installed obviously will not accommodate the full Unit 1 core. Therefore, additional space is needed in the immediate future if Unit 1 is to refuel and continue to operate on schedule.

1.3 Fuel Reprocessing History

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shutdown in 1972 for alterations and expansion; in September, 1976, NFS informed the Commission that it was withdrawing from the nuclear fuel reprocessing business. The Allied

General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York are licensed to store spent fuel. The storage pool at West Valley is not full, but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. GE is also not accepting any additional spent fuel for storage at the Morris Operation.

2.0 The Facility

The principle features of the spent fuel storage and handling at Quad Cities Station as they relate to this action are described here as an aid in following the evaluations in subsequent sections of this environmental impact appraisal.

2.1 The Spent Fuel Pool (SFP)

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core; also, they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal

from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

2.2 SFP Cooling System

The SFP cooling system for each unit at the Quad Cities Station consists of two pumps and two heat exchangers. Each pump is designed to pump 700 gpm (350,000 pounds per hour), and each heat exchanger is designed to transfer 3.5×10^6 BTU/hr from 125 F fuel pool water to 70 F cooling water which flows through the shell side of the heat exchanger.

Heat is transferred from the spent fuel pool cooling system to the reactor building closed cooling water system. The reactor building closed cooling water system, in turn, transfers heat to the service water system. The service water system is a once-through cooling system in which strained water from the Mississippi River is supplied from pumps in the intake structure and returned to the river after removing heat from a number of systems, including the reactor building closed cooling water system.

2.3 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in

the NRC's Final Environmental Statement (FES) dated September, 1972. There will be no change in the waste treatment systems described in Section III.D.2 of the FES because of the proposed modification.

2.4 Spent Fuel Pool Cleanup System

The SFP cleanup system is part of the pool cooling system. It consists of a demineralizer with inlet and outlet filters, and the required piping, valves, and instrumentation. There is also a separate skimmer system to remove surface dust and debris from the SFP. This cleanup system is similar to such systems at other nuclear plants which maintain concentrations of radioactivity in the pool water at acceptably low levels.

3.0 Environmental Impacts of the Proposed Action

3.1 Nonradiological

The nonradiological environmental impacts of Quad Cities Station, as designed, were considered in the FES issued September, 1972. Increasing the number of assemblies stored in the existing fuel pools will not cause any new nonradiological environmental impacts not previously considered. The amounts of waste heat emitted by each of the units as a result of the proposed increased spent fuel storage capacity will increase slightly (less than one percent), but will result in no measurable increase in impacts upon the environment.

3.2 Radiological Consequences of the Proposed Action

3.2.1 Introduction

The potential offsite radiological environmental impact associated with the expansion of spent fuel storage capacity at Quad Cities Station has been evaluated.

During the storage of the spent fuel under water, both volatile and non-volatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54, which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly non-volatile at the temperature conditions that exist in pool storage. The primary impact of such non-volatile radioactive nuclides is their contribution of radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations), or crud dislodged from the surface of the spent fuel during transfer from reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably.

A few weeks after refueling, the spent fuel cools in the pool so that the fuel cladding temperature is relatively cool, approximately 180°F. This substantial temperature reduction reduces the rate of release of fission products from the fuel pellets, and decreases the gas pressure in the gap between pellets and cladding, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on operational reports submitted by licensees, and discussions with storage facility operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

3.2.2 Radioactive Material Released to the Atmosphere

With respect to releases of gaseous materials to the atmosphere, the only radioactive gas of significance which could be attributable to storing additional fuel assemblies for a longer

period of time would be the noble gas radionuclide Krypton-85 (Kr-85). As discussed previously, experience has demonstrated that, after spent fuel has decayed 4 to 6 months, there is no longer a significant release of fission products, including Kr-85, from stored fuel containing cladding defects.

For the simplest and most conservative case, we assumed that all of the Kr-85 that is going to leak from defective fuel will do so in the 18 month interval between refuelings. In other words, all of the Kr-85 available for release is assumed to come out of the fuel before the next batch of fuel enters the pool. Our calculations show that the expected release of Kr-85 from a 200 fuel assembly refueling is approximately 46 Ci each 12 months. As far as potential dose to offsite populations is concerned, this is actually the worst case, since each refueling would generate a new batch of Kr-85 to be released. Since all of the Kr-85 available for release has already left the defected fuel before the next batch enters, the annual releases remain approximately the same. The enlarged capacity of the pool has no effect on the total amount of Kr-85 released to the atmosphere each year. Thus, we conclude that the proposed modifications will not have any significant impact on exposures offsite.

Similarly, Iodine-131 released from stored spent fuel to the pool water will not significantly increase because of the expansion of the fuel storage capacity, since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings for each unit.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 150 F used in the design analysis. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modifications from that previously evaluated in the FES. Most airborne releases of tritium and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the pool water. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increased stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. Charcoal filters are available for the removal of radioiodine from the atmosphere before release to the environment. In addition, the station radiological effluent Technical Specifications, which are not being changed by this action, limit the total releases of gaseous activity.

Based on the foregoing considerations, implementation of the proposed increased spent fuel storage capability will not result in significantly increased amounts of radioactivity being released to the atmosphere.

3.2.2 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The level of activity is highest during refueling operations, when reactor coolant water is introduced into the pool, and decreases as

the pool water is processed through the filters and demineralizer. The increase of radioactivity in the pool water, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the water to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional two resin beds a year, or 160 cubic feet of solid waste, due to the increased operation of the spent fuel pool cleanup system. The annual average volume, per unit, of solid wastes shipped from the Quad Cities Station during 1980 through 1981 was 30,000 cubic feet, so that the 160 cubic feet per unit per year would increase the total waste volume to be shipped offsite by less than 1%. This would have no significant additional environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and might be disposed of as low level solid waste. We have estimated that approximately 7000 cubic feet of solid radwaste will be removed from the plant because of the proposed modification. Averaged over the lifetime of the plant, this would increase the total waste volume shipped from the facility by less than 3%, which we find is not a significant additional environmental impact.

3.2.4 Radioactive Material Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that the change in the quantity and activity of the floor cleanup water as a result of this modification will be insignificant. The SFP demineralizer resin removes soluble radioactive material from the pool water. These resins are periodically sluiced with water to the spent resin storage tank. The amount of radioactivity on the demineralizer resin may increase slightly due to the additional spent fuel in the pool, but the soluble radioactive material should be retained on the resins, to be shipped offsite and buried in sealed drums as solid waste at a licensed burial facility.

Leakage of water from the SFP, if any, would be detected by the pool low level alarm, the flow glass in the drain line and the level detector on the skimmer surge tank. This water would be transferred to the liquid radwaste system for processing and reuse or release to receiving waters.

Based on the foregoing considerations, there will not be a significant increase in radioactivity released to receiving waters as a result of the proposed increase in spent fuel storage capacity.

3.2.5 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks, and the installation of the high density racks, with respect to occupational radiation exposure. The occupational exposure for the operation is estimated by the licensee to be about 18 to 39 man-rem, based on the licensee's detailed breakdown of exposure to each individual performing specific jobs for each phase of the operation. This exposure is a small fraction of the total annual man-rem from occupational exposure for all plant operations.

We have estimated the increase in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of measured dose rates in the SFP area, and from radionuclide concentrations in the SFP water and from the SFP assemblies. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add only a small fraction to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional spent fuel in the SFP will not result in any significant increase in doses received by workers.

3.2.6 Radiological Impacts to the Population

The proposed increase of the storage capacity of the SFP will not create any significant additional radiological effects

to the population. The additional total body dose that might be received by an individual at the site boundary, and by the estimated population within a 50-mile radius, is less than 0.10 mrem/yr and 0.001 man-rem/yr, respectively. These doses are small compared to the fluctuations in the annual dose this population receives from background radiation. The population dose represents an increase of less than 0.01 percent of the dose previously evaluated in the FES for Quad Cities Station. We find this to be an insignificant increase in dose to the population resulting from the proposed action.

3.3 Environmental Impact of Spent Fuel Handling Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated spent fuel handling accident, and a fuel shipping cask drop accident, in the SFP area, from those values previously reported in the Quad Cities FES, based on the following considerations.

The heaviest identified load with this modification is a 16 x 16 rack weighing 16 1/2 tons, whereas the main hoist on the reactor building crane is rated at 125 tons. From a previous review we had concluded that the overhead crane load handling system and the spent fuel cask handling Technical Specifications meet our requirements and are acceptable for handling spent fuel casks weighing up to 100 tons. Spent fuel casks are of course not permitted over spent fuel stored in the pool. The only items transported over spent fuel are other fuel assemblies, pool canal gates,

and a fuel channel measuring device, none of which approach this weight capacity of 125 tons. We have concluded then that the likelihood of a heavy load handling accident is sufficiently small that the proposed modifications are acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are required.

4.0 Summary

The findings contained in the Final Generic Environmental Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, (the FGEIS) issued by the NRC in August, 1979, were that the environmental impact of interim storage of spent fuel was negligible, and the cost of the various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in spent fuel pool designs, the FGEIS recommended licensing spent fuel pool expansions on a case-by-case basis. Expansion of the spent fuel storage capacity at Quad Cities Station does not significantly change the radiological impact evaluated by the NRC in the FES issued in September, 1972. As discussed in Section 3.2.6 of this EIA, the additional total body dose that might be received by an individual at the site boundary or the estimated population within a 50-mile radius is less than 0.10 mrem/yr and 0.001 man-rem/yr. respectively, and is less than the natural fluctuations in the dose this population would receive from background radiation. The occupational exposure for the modifications of the SFPs is estimated by the licensee to be 18 to 39 manrem. This is conservative. Operation of the plant with additional spent fuel in the SFP is not expected to increase the

occupational radiation exposure by more than one percent of the total annual occupational exposure at the two units.

5.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

We have reviewed the proposed modifications relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendments will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an environmental impact statement need not be prepared and that, pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-254 AND 50-265

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO
OPERATING LICENSES
AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 79 to Facility Operating License No. DPR-29, and Amendment No. 73 to Facility Operating License No. DPR-30, issued to Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, which revised the Technical Specifications for operation of the Quad-Cities Nuclear Power Station, Unit Nos. 1 and 2, located in Rock Island County, Illinois. The amendments are effective as of the date of issuance.

The amendments authorize changes to the Technical Specifications to allow an increase in the spent fuel storage capacity from 2920 to a maximum of 7684 assemblies by use of neutron absorbing spent fuel storage racks.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses was published in the FEDERAL REGISTER on April 30, 1981 (46FR47135). Requests for leave to intervene were filed by several citizens groups, and were later withdrawn.

The Commission has prepared an environmental impact appraisal for this action and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action.

For further details with respect to this action, see (1) the application for amendments dated March 26, 1981, as supplemented, (2) Amendment No. 79 to License No. DPR-29, and Amendment No. 73 to License No. DPR-30, (3) the Commission's related Safety Evaluation dated April 9, 1982, and (4) the Commission's Environmental Impact Appraisal dated April 9, 1982. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C., and at the Moline Public Library, 504 - 17th Street, Moline, Illinois. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 9th day of June 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing