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

December 31, 1991

Docket No. 50-333

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M76937)

The Commission has issued the enclosed Amendment No. 175 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated May 31, 1990, and supplemented by letters dated October 31, 1990, December 5, 1990, June 26, 1991, July 12, 1991, July 16, 1991, and September 19, 1991.

The amendment revises the Technical Specifications to allow for the expansion of the spent fuel pool storage capacity from the current 2244 fuel assemblies to the proposed 2797 fuel assemblies. As previously discussed with your staff, during the implementation of this amendment, the NRC staff expects you to adhere to the surveillance and loading requirements specified on pages 4 and 10 of the enclosed Safety Evaluation. Any deviation from these requirements must be reviewed by and have prior approval of the NRC staff.

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

Sincerely,

Brian C. McCabe

Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 175 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

DF01 1/1

DATED: December 31, 1991

AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
~~PDI-1 Reading~~
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
B. McCabe
C. Cowgill
G. Bagchi, 7/H/15
L. Cunningham, 10/D/4
C. McCracken, 8/D/1
C. Y. Cheng, 7/D/4
H. Abelson, 8/E/23
K. Eccleston, 10/D/4
S. Kim, 7/H/15
K. Parczewski, 7/D/4
A. Dummer, 13/E/16
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, P-130A
C. Grimes, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
Plant File

cc: Plant Service list

cc:

Mr. Gerald C. Goldstein
Assistant General Counsel
Power Authority of the State
of New York
1633 Broadway
New York, New York 10019

Resident Inspector's Office
U. S. Nuclear Regulatory Commission
Post Office Box 136
Lycoming, New York 13093

Mr. Radford Converse
Resident Manager
James A. FitzPatrick Nuclear
Power Plant
Post Office Box 41
Lycoming, New York 13093

Mr. J. A. Gray, Jr.
Director Nuclear Licensing - BWR
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Supervisor
Town of Scriba
Route 8, Box 382
Oswego, New York 13126

Mr. John C. Brons, President
Power Authority of the State
of New York
123 Main Street
White Plains, New York 10601

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, New York 10271

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.175
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated May 31, 1990, and supplemented October 31, 1990, December 5, 1990, June 26, 1991, July 12, 1991, July 16, 1991, and September 19, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No.175 , 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 to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 31, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 175

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
246	246
246a	246a

ι΄d)

The spent fuel storage pool is designed to maintain k_{eff} less than 0.95 under all conditions as described in the Authority's applications for spent fuel storage modification transmitted to the NRC July 26, 1978 and May 31, 1990. This k_{eff} value is satisfied if the maximum, exposure dependent, infinite lattice multiplication factor, k_{∞}, of the individual fuel bundle is less than or equal to 1.36. The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 2,797.

.3 SEISMIC DESIGN

The reactor building and all engineered safeguards are designed on a basis of dynamic analysis using acceleration response spectrum curves which are normalized to a ground motion of 0.08 g for the Operating Basis Earthquake and 0.15 g for the Design Basis Earthquake.

Bases

The spent fuel pool and high density fuel storage racks are Class I structures designed to store up to 2,797 fuel bundles. The storage racks are designed to maintain a subcritical configuration having a multiplication factor (k_{eff}) less than 0.95 for all possible operational and abnormal conditions. The nuclear criticality analysis for the Spent Fuel Racks (References 1 and 3) concludes that fresh fuel bundles with 3.3 w/o U-235 meet the 0.95 k_{eff} limit. This design basis bundle was reanalyzed to determine its infinite lattice multiplication factor, k_{∞}, when in a reactor core geometry (Reference 2). This k_{∞} was obtained under conservative calculational assumptions and reduced by 2.33 times the standard deviation in the calculation resulting in the Technical Specification limit of 1.36.

References:

1) Increased Spent Fuel Storage Modification, Stone & Webster Engineering Corporation, Boston, Mass. March 15, 1978.

2) General Electric letter, P. Van Dieman to G. Rorke, FitzPatrick Fuel Storage K-infinity Conversion, Revision 1, dated July 10, 1986.

3) Increased Storage Capacity for FitzPatrick Spent Fuel Pool, Holtec International, Mount Laurel, New Jersey, February, 1989.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 175 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated May 31, 1990, as supplemented October 31, 1990, December 5, 1990, June 26, 1991, July 12, 1991, July 16, 1991, and September 19, 1991, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant, Technical Specifications (TS). The requested changes would revise TS Section 5.5.B and the associated Bases. Specifically, the number of spent fuel assemblies that can be stored in the spent fuel pool (SFP) will be increased from 2244 to 2797.

The FitzPatrick plant SFP was reracked following approval of Amendment No. 55, dated June 18, 1981, with high density racks thus increasing storage capacity to 2244 fuel assemblies. Under the proposed expansion, five new rack modules containing 553 storage locations will be added increasing total storage capacity to 2797. The increased storage capacity will extend the capability for a full-core offload to the year 1997. This effort is consistent with the objective of the Nuclear Waste Policy Act of 1982 which requires that licensees exhaust all means of storing spent fuel on site.

Supplemental letters of October 31, 1990, December 5, 1990, June 26, 1991, July 12, 1991, July 16, 1991, and September 19, 1991, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.A CRITICALITY ANALYSIS

2.A.1 Analytical Methodology

The current Technical Specification (TS) for the spent fuel storage pool and existing racks (TS 5.5.B) states that the k-effective of the pool shall be less than 0.95. The specification further indicates that this k-effective value is satisfied if the maximum exposure dependent k-infinity of the stored fuel assemblies is less than 1.36. The proposed amendment does not change this specification and the pool criterion thus remains at 0.95 for the new racks. The proposed changes to TS 5.5.B are to revise the maximum number of fuel assemblies that may be stored in the pool and to reference the May 31, 1990, submittal. In addition, TS Bases 5.5.B is changed to reflect the new maximum number (2797) of stored fuel assemblies and to eliminate reference to the 3.2% delta-k margin which was calculated for the existing racks but does not apply to the new racks.

The new rack design comprises a rectangular array of stainless steel "box" storage cells. A Boral panel is positioned on each interface between adjoining cells. Each Boral panel is sandwiched between the box wall and a stainless steel sheathing welded to the wall in a manner such that the panel is unconstrained.

The design basis fuel employed in the criticality calculations for the new storage racks is an 8x8 BWR fuel rod assembly with a uniform enrichment of 3.3 w/o U-235 without gadolinium burnable poison. This represents the most reactive fuel authorized for storage at the FitzPatrick facility. For the reference design, the pool moderator is assumed to be pure, unborated water at the minimum temperature within the operating range (68 °F), corresponding to the highest reactivity. The Boron-10 contained in the Boral panels was assumed to be uniformly distributed with a minimum areal density corresponding to the lower limit of the manufacturing tolerance.

The criticality and associated sensitivity calculations were done using both the Monte Carlo code AMPX-KENO (using the 27 group SCALE cross sections, with NITAWL), and, as the primary method, the two-dimensional multi-group transport code CASMO-2E. These methodologies have been benchmarked against a number of relevant critical experiments simulating storage racks designed by Holtec and others. These experiments have covered a range of geometries, material compositions, fuel enrichments, and poison sheets. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k-effective calculations for the FitzPatrick racks.

As part of the sensitivity calculations, the licensee has examined the effect of rack manufacturing tolerances on the computation of k-effective for the rack-fuel system. The parameters considered included boron loading density, Boral panel width, storage cell lattice pitch, and stainless steel box wall and backing plate thicknesses. The effect of storing a fuel assembly without a surrounding zirconium flow channel was also examined. In addition, one-dimensional axial calculations were performed to evaluate the effect of reduced Boral plate lengths on reactivity. Neutron absorption by the structural stainless steel above and below the active fuel was neglected in these computations and fresh unburned fuel with uniform enrichment and no gadolinium was assumed.

The effect of water gap spacing between rack modules was determined by CASMO-2E calculations. The nominal gap along the interface between modules was shown to eliminate the need for Boral panels on the module walls. However, as a precautionary measure against module movement resulting from a seismic event, the rack design provides for Boral panels on the walls of alternate cells along one side of the interface.

The effects on reactivity of abnormal conditions and accidents associated with the spent fuel pool have also been evaluated. These included increased pool water temperature and void formation, the misplacement of a fuel assembly outside and adjacent to the fuel rack, eccentric fuel assembly positioning within a storage cell, lateral rack motion due to a design basis earthquake (discussed above), and a fuel assembly dropped on top of the rack. Computations indicated that these conditions resulted in either a negative reactivity effect or a negligible increase (less than 0.0001 delta-k) in reactivity.

Design basis reactivity calculations resulted in a k-infinity of 0.9297 (bias corrected CASMO). With all known uncertainties statistically combined (a delta-k of ± 0.0071), the maximum k-infinity in the fuel rack becomes 0.937 (95% probability at the 95% confidence level). This satisfies the design basis requirement of a maximum k-effective of less than 0.95. Independent verification calculations using AMPX-KENO resulted in a k-infinity of 0.924 ± 0.008 (95%/95%, corrected for bias and temperature), which is in agreement with the reference calculation.

2.A.2 Conclusion

The basis criticality design of the new racks, using boron lined cells to provide the appropriate neutron multiplication level for the closer packed array of high density racks, is a commonly used concept and has been accepted for many spent fuel storage pools. It is an acceptable design concept for maintaining criticality levels for the FitzPatrick pool.

The analytical methodologies used to analyze the criticality and reactivity change characteristics of the racks are standard methodologies, commonly used and approved for other licensees for such analyses. The CASMO-2E code provides an acceptable methodology for base calculations and for sensitivity calculations, and the AMPX-KENO-SCALE code package provides suitable backup and confirmation calculations. These methods have been benchmarked against an appropriate selection of critical experiments, with results falling within expected ranges of deviations from the experiments. The derivation of the uncertainty of the methodology from this benchmarking follows normal procedures and also falls within an expected range. Therefore, the staff concludes that the criticality analysis is acceptable.

The examination of uncertainties attributed to variances in dimensions and materials in the fuel and racks has covered an acceptable range of parameters and has used a suitable, standard methodology for determining the reactivity effects and their statistical combination. The examination of the effects of abnormal conditions has covered the standard events relating to changes in temperature and density, seismic movements of racks, and misplacement and dropping of fuel assemblies. The staff concludes that these results are acceptable.

2.B MATERIAL COMPATIBILITY AND CHEMICAL STABILITY

2.B.1 Discussion

Nuclear power plants provide storage facilities or pools for the wet storage of spent fuel assemblies. The safety function of the spent fuel storage pools is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. The NRC staff has reviewed the compatibility and chemical stability of the materials wetted in the pool water.

The currently requested expansion would increase the capacity of the spent fuel pool to 2797 fuel assemblies. This expansion will be accomplished by the addition of five new modules. They will be constructed from ASTM A240-Type 304L stainless steel with only the adjustable support spindles made from A564-Type 630 precipitation hardened stainless steel. The neutron absorbing material will be Boral with B-10 loading of 0.0135 gm/sq.cm. Boral is a material consisting of a dispersed boron carbide in a 1100 aluminum alloy matrix and clad with 1100 aluminum alloy. It is in a form of 5 inches wide, 0.075 inches thick, and 144 inches long panels. It is held at the side of the cell by a stainless steel picture frame sheathing. The sheathing is welded to the box at the top and bottom and at staggered positions along the longitudinal length. This design allows pool water free entry to the cavity, and the gases produced by radiolysis and/or water-aluminum reaction are free to escape, thus preventing swelling and bulging due to pressure buildup. The spent fuel pool contains air-saturated demineralized water with conductivity of less than 5μS/cm and chloride contents of less than 500 ppb.

The licensee proposed a surveillance program to monitor performance of the Boral in the spent fuel pool. For that purpose, ten specially designed test coupons will be placed in locations where they will be exposed to the typical spent fuel pool environment. Each coupon will have a Boral specimen encased in a jacket of a material identical to that used in the racks, and the position and tolerances will be similar as that in the actual fuel cell. The jacket will have provisions for easy opening without disturbing the Boral specimen. The coupons will be removed at scheduled intervals and examined for loss of physical and neutron absorbing properties.

2.B.2 Evaluation

The low carbon austenitic stainless steel in the spent fuel racks is compatible with the high purity, demineralized, air-saturated water and the radiation environment of the spent fuel pool. Oxygen dissolved in water will help to passivate the stainless steel. In this environment, austenitic stainless steel will exhibit only extremely low rates of corrosion. These corrosion rates are negligible for even the thinnest stainless steel elements of rack assemblies. Galvanic attack between stainless steel, Zircaloy in the fuel assemblies, and Boral will not be significant since the conductivity of water in the pool is relatively low and the materials are protected by passivating oxide films. The concentration of chloride is maintained below the limit at which significant initiation of stress corrosion cracking could occur.

Boral 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. It has been qualified for 1.0E11 rads of gamma radiation while maintaining its neutron attenuation capability. Tests have shown that Boral does not possess leachable halogens that could be released into the pool environment in the presence of radiation. Similar findings have been made regarding the leaching of elemental boron from the Boral. Surveillance coupons containing Boral will provide time related information of the actual behavior of Boral in the spent fuel pool. The staff reviewed the description of the proposed surveillance program for monitoring the Boral in the spent fuel pool and concludes that the program is adequate to reveal deterioration that might lead to loss of neutron absorbing capability during the life of the spent fuel racks. The staff does not anticipate that such deterioration will occur, but in case it does, it would be gradual. In the unlikely event of Boral deterioration in the pool environment, the monitoring program will detect such deterioration and allow the licensee time to take suitable corrective actions.

2.B.3 Conclusions

Based on the above discussion, the staff concludes that corrosion of the proposed fuel storage racks due to the spent fuel pool environment should be of little significance during the life of the facility. The surveillance program proposed by the licensee would reveal any deteriorations in neutron absorbing capability of Boral and if a significant degradation is found, the licensee would have sufficient time to take the appropriate corrective measures.

The staff finds that the selection of appropriate materials of construction, and the development of a proposed Boral surveillance program meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion 61, regarding the capability to permit appropriate periodic inspection and testing of components, and General Criterion 62 regarding prevention of criticality by the use of neutron absorbers and by maintaining structural integrity of components and are, therefore, acceptable.

2.C THERMAL/MECHANICAL LOAD CONSIDERATIONS

2.C.1 Spent Fuel Pool Cooling

No modifications to the spent fuel pool (SFP) cooling system are necessary with the proposed expansion. Therefore, the spent fuel pool cooling system was only reviewed against the requirements of General Design Criterion (GDC) 44 for decay heat removal and GDC 2 for makeup during loss of all cooling as defined in Standard Review Plan, Section 9.1.3, for storage of 2797 fuel assemblies.

The FitzPatrick spent fuel pool cooling system consists of two pumps and two heat exchangers for normal decay heat removal. Both heat exchangers, when supplied by a single SFP cooling pump, are designed to transfer 6.3×10^6 BTU/hr from 125 °F fuel pool water to 95 °F reactor building closed loop cooling water which flows through the shell side of each heat exchanger. The spent fuel pool cooling system can be supplemented by the use of the RHR system in the spent fuel cooling assist mode. The RHR assist mode is available during plant shutdowns. When the temperature of the spent fuel pool exceeds the peak efficiency temperature of 100 °F for the spent fuel pool cleanup system, the filters and demineralizers can be bypassed.

The licensee calculated decay heat loads of 13.1×10^6 BTU/hr after a normal discharge of spent fuel during refueling. This is based on a proposed storage capacity of 2797 spent fuel assemblies. The heat load value was compared to Branch Technical Position 9-2 and found to be conservative. The calculated pool temperature rises to a maximum of less than 150 °F at 165 hours after shutdown for a normal discharge of 208 fuel assemblies. For a single active failure after the plant has started up, the maximum fuel pool temperature would be maintained less than 150 °F. This maximum temperature is above the guideline of 140 °F for a normal discharge; however, it is acceptable because it is well below the boiling temperature.

When a full core is offloaded into the spent fuel pool, the Residual Heat Removal (RHR) System will be used to maintain the fuel pool temperature at or below 135 °F. The use of the RHR assist mode for cooling when the full core is unloaded was accepted in a Safety Evaluation dated June 18, 1981. The decay heat load for a full-core offload is calculated to be 25.79×10^6 BTU/hr at 238 hours after shutdown. This heat load results in a calculated maximum temperature of 133 °F with RHR assist in operation. The maximum temperature of the pool for the abnormal condition of full-core offload with RHR assist is acceptable because if is below the boiling temperature.

Makeup for the SFP is manually transferred from the seismic Category I condensate storage system to the skimmer surge tanks to make up any pool losses. Capability exists to add water from Lake Ontario to the pool through the RHR system in the event of loss of normal makeup system and when pool water level is threatened due to heavy pool water inventory loss.

Based on the above, the decay heat removal for normal and abnormal conditions, and the makeup capability are acceptable.

2.C.2 Heavy Load Handling

A spent fuel storage rack is considered to be a heavy load because is weighs more than a spent fuel assembly and its handling tool. The licensee indicated that lifting and installation of the spent fuel racks will be performed in accordance with the guidelines of NUREG-0612. All load handling will follow clearly established safe load handling paths. The crane operator will be given special training and will be required to follow specific load handling procedures. The lifting crane and the rig will meet the NUREG-0612 stress and inspection criteria. In a Safety Evaluation dated January 3, 1984, the licensee's provisions for handling and control of heavy loads at the FitzPatrick Nuclear Power Plant were found to meet the guidelines of NUREG-0612.

Based on the above, the staff finds that heavy load handling will be performed in accordance with the guidelines of NUREG-0612 to ensure that an unacceptable release of radioactivity or criticality accident will not result from a heavy load drop, and is therefore acceptable.

2.D STRUCTURAL DESIGN

This evaluation addresses the adequacy of the structural and seismic aspects of the application submitted by the licensee in support of their increased rack capacity in the spent fuel pool. The primary areas of review are focused on the structural integrity of the fuel, fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated loads (Appendix D of SRP 3.8.4) and fuel handling accidents.

2.D.1 Structural Analysis

Spent Fuel Storage Pool

The spent fuel pool is a reinforced concrete structure and is designed as a Seismic Category I structure. The pool is approximately 31 feet wide, 40 feet long, and 37 feet deep with a 5 feet thick slab. Wetted surfaces of the pool are lined with stainless steel to ensure water tight integrity.

The concrete strength capacities were compared with anticipated loads on the concrete structure from high density rack dynamic loads as well as other loadings specified in the Standard Review Plan and the margins were found to be acceptable.

The staff, therefore, concludes that the FitzPatrick spent fuel pool will continue to support the additional loads caused by additional fuel during normal, severe environmental, and accident conditions and maintain its integrity.

Refueling Accidents

The following three accidents were evaluated by the licensee: (a) a fuel assembly is dropped from an elevation 24" above a storage location and impacts the base of the module, (b) a fuel assembly is dropped from an elevation 24" above the rack and hits the top of the rack, and (c) the same as (b) except that the fuel assembly is assumed to be dropped in an inclined manner on the top of the rack. The licensee found that the above postulated accidents would not lead to adverse conditions, including unacceptable damage to the fuel. Furthermore, the licensee found that the rack cross-sectional geometry would not be altered during these accidents.

The NRC staff has evaluated the licensee's analyses and concludes that its findings are acceptable.

Rack Modules

The five racks to be added to the pool are seismic Category I equipment and are, therefore, required to remain functional during and after a Safe Shutdown Earthquake (SSE). They are neither anchored to the pool floor nor the pool wall and are not structurally interconnected. Each rack module is provided with leveling pads which support the rack and are in contact with the spent fuel pool floor.

The licensee performed dynamic analyses that concluded that the rack modules would not develop enough kenetic energy during a SSE to damage the spent fuel pool liner or the rack modules themselves. Furthermore, analysis performed by the licensee also found that there will be no rack-to-rack, or rack-to-pool wall impact during a SSE.

Racks C1 and C2 have more moving space than the other three racks. For those two racks, potential for a tip over is greater than for translational movement. The licensee's calculation, based on the DYNARACK code, indicated that such a possibility does not exist. However, the code was benchmarked to an incomplete verification process and the theoretical aspect of the highly nonlinear calculation has not been documented adequately to address the staff concerns on potential numerical error and instability. The staff, for this reason, did not solely rely upon the results of the DYNARACK analysis to make a safety evaluation of the racks. The licensee indicated that their vendor, Holtec, has performed an experiment that demonstrated that rack-to-wall impact is unlikely. However, the licensee has not documented this experiment formally and, consequently, the staff has not reviewed the basis of the experimental findings. Therefore, the staff made the following independent assessment to supplement licensee's calculation and design adequacy conclusion.

Based on the geometry and aspect ratio of the racks, the NRC staff has determined that, were one to tip over, it would be more likely to occur in the east-west direction that the north-south direction. Simplified assessment of the rack lift-off potential, based on rack overturning stability considerations in conjunction with a conservative use of the floor response spectrum at elevation 236 feet, indicates that there is a possibility of one side of the supporting legs to lift off the pool floor. The acceleration needed to lift the rack in the east-west direction was found to be approximately 0.2g (horizontal) when a combined excitation by the full horizontal acceleration and two-thirds of the horizontal acceleration as the vertical input was assumed to act on the rack simultaneously, and no resistance from water against the lifting was considered. The staff also evaluated the actual safety margin against overturning of the rack. It was found that the conservatively assessed safety factor against overturning the rack is approximately 1.1, which is consistent with the provision of SRP Section 3.8.5, and, therefore, is acceptable. This evaluation was based on the conservation of energy principle whereby the kinetic energy resulting from the maximum velocity of the rack induced by an earthquake is equated to the potential energy that is needed to raise the rack to position where the center of gravity of the rack moves beyond a line connecting the two supporting legs of the rack. Rack stresses due to the horizontal inertial force corresponding to the SSE were also found to be small and acceptable. However, for increased safety margin, the NRC staff requires that racks C1 and C2 be either kept empty or loaded in such a way that the center of gravity of a partially loaded rack be maintained at least a distance of one half the rack width in the east-west direction away from the rack boundary which is closest to the pool wall (i.e., minimize the potential for rack tip toward the east pool wall).

Fuel handling equipment, specifically a channel storage rack and fuel preparation machines, occupies part of the space within the distance between the racks and the pool wall. Based on the proximity and dimensions of this equipment, the NRC staff has concluded that in the unlikely event of a rack tip-over during an earthquake and subsequent impact with this equipment, the rack would not develop sufficient kinetic energy to damage itself to such an extent that the basic fuel assembly integrity would be compromised.

Finally, the FitzPatrick plant is located in a low-seismic-activity zone. The NRC staff believes that ground motion capable of leading to significant dynamic excitation of the rack is highly unlikely.

2.D.2 Conclusion

Based on the review and evaluation of the licensee's submittals, and the staff's independent assessment, it is concluded that the spent fuel rack modules and the spent fuel pool are capable of withstanding the abnormal loading associated with a SSE in combination with other applicable loads. Furthermore, the design of the spent fuel modules and the spent fuel pool are in conformance with the applicable acceptance criteria established in the Standard Review Plan and are consistent with the current licensing practice. Therefore, the NRC staff concludes that the structural aspects of the additional racks are acceptable.

Maintenance of uniform gaps between the racks and between the racks and pool wall is desirable from a structural point of view since it minimizes a potential for impact. Therefore, the staff requires the licensee to institute a surveillance program that inspects and maintains rack gaps after an earthquake equivalent to or larger than an Operating Basis Earthquake (OBE), if any occurs. The surveillance should also include inspection of rack and fuel integrity for any damage. In addition, the staff requires that racks C1 and C2 be either kept empty or loaded in such a way that the center of gravity of a partially loaded rack be maintained at least a distance of one half the rack width in the east-west direction away from the rack boundary which is closer to the pool wall.

2.E RADIATION PROTECTION AND ALARA CONSIDERATIONS

2.E.1 Occupational Exposure Controls

The Spent Fuel Pool rack addition will fall under the responsibility of the FitzPatrick Radiological and Environmental Services Department and will require pre-job briefings, man-rem estimates, and exposure tracking. Radiation, contamination, and airborne surveys will be performed prior to any work in the pool, and radiological conditions along with protective clothing requirements will be stated on the applicable Radiation Work Permits.

Storing additional spent fuel in the pool will increase the amount of corrosion and fission product radionuclides introduced into the pool water. Specifically, activated corrosion products such as Co-58, Co-60, Fe-59, and Mn-54 may be released to the pool from the surface of the spent fuel assemblies and fission products such as Cs-134, Cs-137, Sr-89, and Sr-90 may be released to the pool water through defects in the spent fuel cladding. However, the additional activity introduced to the fuel pool from the increase in stored fuel assemblies should not increase radiation dose rates above the fuel pool. Furthermore, the spent fuel stored in the new racks will be shielded by approximately 24 feet of water resulting in negligible dose rates above the fuel pool.

The collective occupational dose for the proposed modification of the SFP is estimated by the licensee to be about 2 person-rem. Based on previous experience with related activities at similar facilities, the staff believes that the licensee's estimate is low and that collective doses for these activities will more likely fall in the range of 4-6 person-rem.

The licensee has indicated that the removal of irradiated material currently stored in the spent fuel pool where the additional racks will be installed is estimated to require collective doses of about 13.5 person-rem. The licensee has further stated that this 13.5 person-rem is not directly attributed to the new rack installation. Even if this exposure were included in its entirety, and the staff value of 4-6 person-rem were used to estimate occupational radiation exposures for the rack installation, the total additional collective dose of 17.5-19.5 person-rem is a small fraction of the 1987-1989 average annual occupational dose for FitzPatrick. This small increase in collective radiation dose should not affect the licensee's ability to maintain individual occupational doses within the limits of 10 CFR Part 20, and is as low as is reasonably achievable. Normal radiation control procedures should preclude any significant occupational exposures.

Based on present and projected operations in the SFP area, we estimate that the proposed expansion of the SFP should add less than 3% to the total annual occupational radiation dose at the facility, based on the average collective dose reported by the licensee for the 1987-1989 period.

Therefore, we conclude that the proposed storage of additional fuel in the modified SFP will not result in any significant increase in doses received by workers.

2.E.2 Accident Analyses

The staff, in the Safety Evaluation Report issued March 4, 1970, addressed the safety and environmental aspects of a fuel handling accident. A fuel handling accident may be viewed as a "reasonably foreseeable" design basis event which the pool and its associated structures, systems, and components (including the racks) are designed and constructed to prevent. The environmental impacts of the accident were found not to be significant.

The staff has reviewed the accidental fission product releases that could occur at FitzPatrick in conjunction with the proposed expansion of the spent fuel storage capacity. The staff finds that neither the reracking operations nor the increased capacity of spent fuel storage resulting from the proposed modification affect the calculated consequences of postulated accidents. Likewise, the proposed rack addition does not create the possibility of a new type of accident not previously analyzed. The radiological consequences resulting from postulated accidents have been previously analyzed and found acceptable as specified in the applicable regulation at 10 CFR Part 100.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact have been prepared and published (56 FR 66460) in the Federal Register on December 23, 1991. 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 published a Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for Hearing in the Federal Register on July 24, 1990 (56 FR 30051). No requests for hearing were received and the State of New York did not have any comments.

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
H. Abelson
K. Eccleston
S. Kim
K. Parczewski
A. Dummer

Date: December 31, 1991

December 31, 1991

Docket No. 50-333

DISTRIBUTION:
See attached sheet

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M76937)

The Commission has issued the enclosed Amendment No. 175 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated May 31, 1990, and supplemented by letters dated October 31, 1990, December 5, 1990, June 26, 1991, July 12, 1991, July 16, 1991, and September 19, 1991.

The amendment revises the Technical Specifications to allow for the expansion of the spent fuel pool storage capacity from the current 2244 fuel assemblies to the proposed 2797 fuel assemblies. As previously discussed with your staff, during the implementation of this amendment, the NRC staff expects you to adhere to the surveillance and loading requirements specified on pages 4 and 10 of the enclosed Safety Evaluation. Any deviation from these requirements must be reviewed by and have prior approval of the NRC staff.

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

Sincerely,
ORIGINAL SIGNED BY:
Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 175 to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

OFC	:PDI-I:LA	:PDI-I:PM	:OGC*	:PDI-I:D	:
NAME	:CVogan CV	:BMcCabe:avl BM	:	:RCapra RC	:
DATE	:12/31/91	:12/31/91	:12/9/91	:12/31/91	:

OFFICIAL RECORD COPY
Document Name: FITZ AMDT M76937

~~9201080140~~ 3pp