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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 256 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

On October 14, 1997, as supplemented on July 23, 1998, December 3, 1998, February 25, 1999, and September 29, 1999, the Power Authority of the State of New York (the licensee, also known as the New York Power Authority) proposed changes to the Technical Specifications (TSs) for the James A. FitzPatrick Nuclear Power Plant (JAF). The proposed changes consist of revisions necessary to permit use of additional racks for fuel storage in the spent fuel pool (SFP) to increase the pool capacity from 2797 to 3239 fuel assemblies, and to change the infinite lattice multiplication factor to ensure fuel in the pool is maintained in a subcritical configuration. The additional information provided in the July 23, 1998, December 3, 1998, and February 25, 1999, supplements did not affect the staff's proposed finding of no significant hazards consideration, and was within the scope of the amendment application as noticed.

2.0 EVALUATION

The licensee proposes changes to TS section 5.5, "Fuel Storage." The proposed changes completely replace this section as follows:

5.5 FUEL STORAGE

5.5.1 Criticality

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

a. Fuel assemblies having a maximum k_∞ of 1.32 in the normal reactor core configuration at cold conditions (20 °C);

b. $k_{eff} < 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.3 of the FSAR; and

c. A nominal center to center distance between fuel assemblies placed in the storage racks as described in Section 9.3 of the FSAR.

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

a. Fuel assemblies having a maximum k_∞ of 1.31 in the normal reactor core configuration at cold conditions (20 °C);

b. $k_{eff} \leq 0.90$ if dry;

c. $k_{eff} \leq 0.95$ if fully flooded with unborated water; and

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

5.5.2 Capacity

The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 3239 fuel assemblies.

The licensee also proposes format and editorial changes to TS Sections 5.1, 5.2, 5.3, and 5.4. TS 5.1.A is proposed to be renumbered TS 5.1.1, and the word "country" replaced by "county." This change corrects an editorial error in the existing specification. The licensee also proposes to renumber TS 5.1.B, 5.2.A, 5.2.B, 5.4.A, 5.4.B, and 5.4.C as 5.1.2, 5.2.1, 5.2.2, 5.4.1, 5.4.2, and 5.4.3, respectively.

The NRC staff's evaluation of these changes is presented below.

2.1 HEAVY LOADS

The installation and use of the additional fuel racks involves the handling and control of heavy loads. Certain components are considered heavy loads because they weigh more than a spent fuel assembly and its handling tool. This review is focused on removal and installation of the spent fuel storage racks, the gate between the reactor canal and SFP, the design and use of the hoisting system, procedures, crane operator training, and postulated load drop accident analyses and consequences.

2.1.1 Hoisting System

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" provides guidelines for licensees to (1) assure safe handling of heavy loads, (2) reduce the potential for uncontrolled movement of heavy loads or load drops, and (3) limit the consequences of load drops over spent fuel, fuel in the reactor core, and safety-related equipment.

The licensee proposes to use a "defense in depth" approach in accordance with NUREG-0612, Phase I to move heavy loads during the SFP reracking activities at JAF. Accordingly, the licensee plans to use redundantly designed equipment, trained personnel, and comprehensive control procedures to satisfy guidelines recommended in NUREG-0612. A previous licensee's 10 CFR Part 50.59 Safety Evaluation, JAF SE-97-003, Rev. 1, that was found acceptable to the staff (NRC Integrated Inspection Report No. 50-333/98-02 dated July 2, 1998), stated that the Reactor Building overhead crane will be used to move spent fuel assemblies in the SFP. A remotely controlled lifting rig will be used with the Reactor Building overhead crane to handle the spent fuel racks. The lifting rig is specifically designed to handle the spent fuel racks. It is designed and tested in accordance with the guidelines in NUREG-0612 and requirements in ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." The 125-ton crane is designed in accordance with the requirements of CMAA No. 70, "Specifications for Electric Overhead Traveling Cranes." The lifting rig consists of redundant components such that failure of a single component will not result in uncontrolled lowering of the rack. Both the stress design and the load testing of the lifting rig satisfies guidelines in Section 5.1.6(1)(b) of NUREG-0612 and ANSI N14.6 (1978), respectively. As stated by the licensee, the lifting rig is designed as follows: (1) with twice the normal stress design factor (a safety factor of 10 to 1); (2) load tested to three times the maximum weight to be lifted with the load sustained for a minimum of 10 minutes; and (3) after load testing, the integrity of the critical joints and welds are examined using liquid penetrant. Other lifting devices used by the licensee during the rerack operation are designed and tested in accordance with ANSI B30.9 - 1971, "Slings."

Further, to enhance its load lifting capability, the licensee stated that they will undertake additional measures that are in accordance with NUREG-0612 to help maintain safety during the installation of the new racks. These measures involve the use of administrative controls and procedures to preclude load drop accidents based on the four major causes of accidents cited in NUREG-0612: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee plans to provide comprehensive training to the rack installation crew, use redundantly designed lifting rigs, use rack inspection procedures to avoid rework, and use procedures throughout the rerack process.

The NRC staff believes that the crane, lifting rig, and the additional measures taken by the licensee following guidelines in NUREG-0612 will ensure safe lifting of the racks, the gate and other loads during the rerack operation. The design and testing of the lifting rig and other lifting devices, and the limit on lifts made with the overhead crane will enable the licensee to handle heavy loads without any risks to the safety of the rerack operation.

2.1.2 Postulated Load Drop Accidents

The licensee presented a refueling accident analysis that evaluated the consequences of dropping a spent fuel assembly, a spent fuel rack, and the gate between the reactor canal and SFP. A drop of a spent fuel assembly from approximately 24 inches above the spent fuel storage cells would not result in any significant safety impacts: the top region of the fuel would have minor damage but the subcriticality of the fuel would not change; the top of the racks would have limited deformation that is acceptable; the SFP liner would not be affected, and release of radioactive material would not occur.

Licensee analysis of postulated drops of both a spent fuel rack and the gate onto existing racks and the SFP liner shows that the consequences would be negligible. Dropping a rack onto spent fuel assemblies is highly unlikely because the rack modules will be empty, and at least 2 feet laterally from a loaded storage cell when it is hoisted beyond 6 inches above the pool floor. Dropping the gate onto the racks would cause minor deformation of the racks with negligible impact on the top region of the spent fuel.

The licensee stated that the case of a heavy load drop onto the SFP liner was previously evaluated in a safety evaluation (SE) dated January 3, 1984, and was found to meet the guidelines in NUREG-0612. This is also noted in an NRC SE dated December 31, 1991. Based on these evaluations, the staff concludes that the SFP would not experience any damage that would result in water leakage and uncover the fuel, and the potential for damaging safe-shutdown equipment is highly unlikely.

In addition, the licensee proposed to implement administrative controls and procedures to preclude load drop accidents. These include (1) performing preventive maintenance and inspections of the crane at least 3 months prior to the rerack, (2) limiting the lifts to less than 50% of the crane's capacity, (3) establishing safe load paths that avoid transport of the racks over fuel in the pool or in the proximity to safety-related equipment, and (4) providing training for the rerack crew. The staff agrees with the licensee that its plans to use administrative procedures and controls focused on the areas noted above would enhance the safety of the rerack operation.

2.2 SPENT FUEL POOL COOLING

SFP cooling was originally provided by the fuel pool cooling and cleaning (FPCC) system and the residual heat removal (RHR) system in the SFP cooling assist mode. Prior to the Cycle 12 refueling outage in 1996, JAF installed an independent decay heat removal (DHR) system to remove decay heat from the core and the SFP. The DHR system allows surveillance testing, modifications, maintenance and repairs to be performed on the RHR, FPCC and service water systems as early as possible during a refueling outage, providing greater flexibility in refueling outage planning. With the addition of the DHR system, three systems are available at JAF to maintain the SFP below its design temperature of 150 °F under a variety of heat loads and off-load scenarios.

The FPCC system, which is a nonsafety-related and non-seismic designed system, consists of two 100% capacity pumps and two 50% capacity heat exchangers. The FPCC system is

designed to transfer a heat load of 6.3 x 10^6 Btu/hr from 125 °F SFP water to 95 °F reactor building closed loop cooling water with one pump and two heat exchangers operating in parallel. For the current SFP capacity after the final discharge of 208 spent fuel assemblies (partial core), the maximum SFP heat load will be 13.1 x 10^6 Btu/hr. The FPCC system operating with two pumps and two heat exchangers will maintain the peak SFP water temperature below 146 °F with this decay heat load.

The RHR system, which is a safety-related and seismic Class 1 designed system, consists of four pumps and two heat exchangers. Permanent connections with normally closed valves are provided in the shutdown cooling piping circuit for supplying cooling water to the FPCC system. Any one of four RHR pumps can be aligned to one of the two RHR heat exchangers in the SFP cooling assist mode. In the event that the SFP heat load exceeds the heat removal capability of the FPCC system, the RHR system provides supplemental cooling to the SFP. For a full core off-load at the current SFP capacity, the decay heat load will be 25.79 x 10^6 Btu/hr. The combined FPCC system and the RHR system in SFP cooling assist mode have sufficient cooling capacity to maintain the peak SFP water temperature below 133 °F with this decay heat load.

During refueling outages, after the reactor pressure vessel head has been removed, the reactor cavity has been flooded, and the fuel transfer gates have been removed, the DHR system can be placed in service to supplement or substitute for the functions provided by the FPCC system and the RHR system in its SFP cooling assist mode.

The DHR system is comprised of a primary loop which pumps water from the SFP through heat exchanger(s) and returns it to the SFP and a secondary loop which removes the heat from the primary loop heat exchangers via cooling towers. The primary loop consists of two pumps, two heat exchangers and associated piping, valves and instrumentation controls. The secondary loop consists of two pumps, two sets of cooling towers (two cooling towers make a set), piping, valves and controls. The DHR system is designed to provide a nominal heat removal capability of 30 x 10^6 Btu/Hr at SFP water temperature of 125 °F when operating the primary loop with one pump and one heat exchanger and the secondary loop with one pump and one set of cooling towers. In the design maximum heat removal configuration (using one primary loop pump, two primary loop heat exchangers, two secondary loop pumps and one set of cooling towers), the DHR system heat removal capability is 45 x 10^6 Btu/Hr at SFP water temperature of 125 °F.

The DHR system is a non-safety related and non-seismic designed system. It is powered from a reliable offsite power source (existing 13.2 KV switchgear J02) which is independent of the existing safety-related and non-safety-related power supplies to the power block. During refueling outages, DHR system reliability will be enhanced through the use of a portable (truck-mounted) diesel generator. In addition, selected spare DHR system components including spare primary and secondary pumps will be maintained available on site during refueling outages to ensure that the DHR system could be restored to operation in a timely manner in case of system problems.

The design heat removal capabilities of the SFP cooling systems for various configurations with SFP water temperature of 125 °F are:

System Configuration	Heat Removal Capability, MBTU/hr
DHR (1 primary pump, 2 primary heat exchangers + 2 secondary pumps, 2 sets of cooling towers)	45
DHR (1 primary pump, 1 primary heat exchanger + 1 secondary pump, 1 set of cooing towers)	30
RHR Assist + FPCC (1 pump, 1 heat exchangers)	24
FPCC (2 pumps, 2 heat exchangers)	10
FPCC (1 pump, 2 heat exchangers)	6.3

With the DHR system available or the RHR system maintained in the SFP cooling assist mode, SFP cooling can tolerate the failure of an active component with no degradation of decay heat removal function. In support of this capability, the plant implemented administrative controls and operating procedures when the DHR was installed to ensure that 100% backup cooling capability is provided for all SFP cooling scenarios. These controls and procedures will remain in place for the proposed increased SFP storage capacity.

Since the proposed increase in SFP storage capacity would result in an increase of SFP heat load for any specific fuel discharge scenario, the licensee re-evaluated the effects of increased SFP storage capacity on the SFP heat loads and temperatures. The following table compares the SFP heat loads and their corresponding peak SFP temperatures for the current SFP storage capacity of 2797 spent fuel assemblies (SFA), and the proposed SFP storage capacity of 3239 SFAs during partial or a full core off-load[1]:

Case	In-core Hold Time (hrs)	Peak SFP Temp. (°F)	SFP Heat Load (MBtu/hr)
1A	96	146	13.1
1B	96	147.64	13.61
2A	96	133	25.79
2B	96	143.83	31.12

Case 1A: Partial Core Off-load - Based on: an SFP capacity of 2797 SFAs, 18 month fuel cycle, off-load 208 SFAs at a rate of 4 SFAs per hour, and cooling the SFP with the FPCC system using 2 pumps and two heat exchangers.

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[1] Partial core (208 fuel assemblies) discharge is the routine practice for refuelings. Full core (560 fuel assemblies) discharge is an option for refueling operation at JAF.

Case 1B: Partial Core Off-load - Based on: an SFP capacity of 3239 SFAs, 24 month fuel cycle, off-load 208 SFAs at a rate of 4 SFAs per hour, and cooling the SFP with the FPCC system using 2 pumps and two heat exchangers.

Case 2A: Full Core Off-load - Based on: an SFP capacity of 2797 SFAs, 18 month fuel cycle, 30 days after the final discharge of 144 SFAs plus a full core off-load at a rate of 4 SFAs per hour, and cooling the SFP with the FPCC system and RHR system in the SFP cooling assisting mode.

Case 2B: Full Core Off-load - Based on: an SFP capacity of 3239 SFAs, 24 month fuel cycle, 30 days after the final discharge of 208 SFAs plus a full core off-load at a rate of 4 SFAs per hour, and cooling the SFP with the FPCC system and RHR system in the SFP cooling assisting mode.

As shown in the above table for partial core off-load, the SFP decay heat load will increase from 13.1×10^6 Btu/hr to 13.61×10^6 Btu/hr and the corresponding peak SFP temperature increases from 146 °F to 147.64 °F. For a full core off-load, the SFP decay heat load will increase from 25.79×10^6 Btu/hr to 31.12×10^6 Btu/hr and the corresponding peak SFP temperature increases from 133 °F to 143.83 °F. The peak SFP temperatures resulting from the proposed SFP storage capacity are acceptable because they are below the SFP design temperature of 150 °F.

Since the design heat removal capabilities of the DHR system are higher than the combined design heat removal capabilities of the FPCC system and the RHR system in the SFP cooling assist mode, the licensee has not performed analyses to evaluate the effects of the increased SFP storage capacity on the DHR system.

The staff finds that the design and operation of the SFP cooling systems (FPCC system, RHR system in the SFP cooling assist mode and DHR system) meet the intent of the guidance described in Standard Review Plan (SRP) for SFPs. The staff also concludes that the SFP will be maintained below its design temperature of 150 °F for normal offloads (partial core) through emergency full core offloads (full core offload 30 days following a normal offload).

2.2.1 Effects of SFP Boiling

In the unlikely event that there is a complete loss of cooling, the SFP water temperature will begin to rise and eventually will reach the boiling temperature.

The licensee performed analysis to demonstrate the time to boil and the boil-off rate. This analysis was performed using the TBOIL computer program. The licensee's original submittal provided results for 2 decay heat load cases. For the more limiting case, the calculated minimum time from the loss-of-pool cooling at peak pool water temperature until the pool boils based on the heat load for the full core off-load is 5.75 hours with 38.3 gpm makeup required to prevent pool level from dropping to less than 10 feet above the stored fuel.

The staff noted that the licensee's December 3, 1998, submittal revision included changes to the results for this limiting case, reducing the time-to-boil to 5.04 hours, and increasing the required makeup flow to 47.3 gpm. In response to a January 26, 1999, request for additional information, the licensee stated that the changes were the result of an error in the TBOIL decay heat calculation which had been identified and corrected prior to the December 3, 1998 submittal.

The licensee provided information on the mass of water to be heated within the pool, initial conditions, and the decay heat load. The staff performed a confirmatory calculation estimating the time to boil as 5.08 hours. The estimated makeup flow was found to be about 57 gpm to fully compensate for pool boil-off. The lower value provided by the licensee reflects the minimum flow required to prevent level from dropping to less than 10 feet above the stored fuel, and reflects the change in decay heat load with time.

The licensee stated that makeup water for SFP is obtained from the Seismic Class 1 condensate storage tanks by manual operation of the condensate transfer system. There are also provisions to add lake water to the pool through the RHR system in the event of loss of normal makeup system. In addition, the fire protection system with two diesel-driven pumps is another source of water supply. Each of these sources can provide make-up water at a rate that ensures pool level does not drop excessively.

The staff finds that in the unlikely event of a complete loss of cooling, the licensee is capable of aligning makeup water from various sources to the pool before boiling begins. The staff concludes that cooling the SFP at JAF by adding makeup water during an unlikely event of a complete loss of SFP cooling conforms with the guidance described in the SRP, and is, therefore, acceptable.

2.3 STRUCTURAL INTEGRITY

The staff evaluated the proposed rack installation to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structure subject to the effects of the postulated loads (per NUREG-0800, Section 3.8.4, Appendix D) and fuel handling accidents.

2.3.1 Storage Racks

The 3239 spent fuel assemblies will be contained in 38 fuel storage racks, which are seismic Category I equipment and are required to remain functional during and after a safe shutdown earthquake (SSE). The licensee's original submittal proposed additional capacity of 442 assemblies for a total pool capacity of 3247 assemblies. The capacity was reduced when it was found that one of the racks would not fit in the pool as originally designed, and this rack was modified by removal of a row of storage cells. The discussion below addresses first the structural analysis submitted on October 14, 1997, with additional information provided on July 23, 1998, then structural analysis effects resulting from the reduced pool capacity, which were described in the licensee's December 3, 1998 submittal.

2.3.1.1 Original Storage Rack Structural Analysis

The NRC previously licensed 31 racks for reracking Campaigns I and II and they are in service at FitzPatrick. The licensee is proposing to add 7 new high density storage racks to the pool to accommodate the increase of 450 (later revised to 442) fuel assemblies for the reracking Campaigns III and IV. The licensee with its contractor, Holtec, performed structural analyses for 7 new racks for the requested license amendment.

The computer program, DYNARACK, was used for dynamic analysis to demonstrate the structural adequacy of the FitzPatrick spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor of the storage pool. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, were used to simulate three dimensional dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

A series of 3-D single rack (SR) model analyses was performed. Four rack geometries were considered in the SR analysis for the calculation of stresses and displacements:

1. 4.3 ft (W) x 6.9 ft (L) x 15.2 ft (H),

2. 2.1 ft (W) x 4.8 ft (L) x 15.2 ft (H),

3. 2.1 ft (W) x 2.1 ft (L) x 15.2 ft (H) and

4. 3.2 ft (W) x 8.0 ft (L) x 15.0 ft (H),

where W, L and H are defined as width, length and height of a rack, respectively. Each rack was considered fully loaded, half loaded and almost empty with three different coefficients of friction between the rack and the pool floor (μ=0.2, 0.8 and random) to identify the worst-case responses for rack movement and for rack member stresses and strains.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration time histories) was generated in conformance with the design response spectra defined in the final safety analysis report (FSAR). The licensee demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying the requirements of both enveloping the design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in the Standard Review Plan (SRP) Section 3.7.1.

The 3-D SR analyses were performed under the service, upset and faulted loading conditions (Level A, B and D Service Limits). The results of the analyses show that the maximum displacements of the racks at the top and the baseplate corners are about 0.2 inch and 0.02 inch, respectively, indicating that there is adequate safety margin against overturning of the racks and, thereby, the structural integrity and stability of the racks is maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Subsection NF. The results show that all induced stresses under SSE loading conditions are smaller than the corresponding allowable stresses specified in the ASME Code indicating that the rack design is adequate.

The licensee did not perform a 3-D multi-rack (MR) analysis. In a request for additional information dated June 1, 1998, the licensee was requested to justify not performing a 3-D MR analysis. It is the staff's understanding that, in general, a 3-D MR analysis provides more critical information for evaluating structural integrity of racks than a 3-D SR analysis does.

On July 23, 1998, the licensee justified not performing a 3-D MR analysis for the following reasons:

1. evaluations performed for the last storage capacity expansion project (Campaign II) accepted by NRC showed that the SR analyses bounded similar analyses performed using the MR technique,

2. the currently proposed 7 new racks are not all located contiguously in the pool but are located adjacent to the 31 existing racks, and

3. the weak seismic response (e.g., almost no rack movement) and low stresses in the 7 racks are found during an SSE from the SR analyses.

In addition, the licensee indicated that the SR analyses were comprehensive in-depth, detailed analyses that were carried out with the following parameters: (1) a large number of 3-D SR simulations (total of 104), (2) full, half-full and almost empty loading configurations within each rack, (3) three different coefficients of friction between the rack and the pool floor (μ=0.2, 0.8 and random), and (4) considerations of in-phase and opposed-phase of adjacent racks. In view of the NRC's previous acceptance on the results of the licensee's comparison study between the SR and MR analyses for Campaign II, almost no rack movement (less than 0.2 inch) of the proposed 7 new racks, thereby, no impact between rack to rack and rack to wall during an SSE, very low induced stresses in the racks during an SSE, and comprehensive SR analyses with parameters delineated above, the staff agrees that the licensee's justification for not performing a 3-D MR analysis is acceptable.

The licensee calculated the weld stresses for the rack at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all the calculated weld stresses are smaller than the

corresponding allowable stresses specified in the ASME Code indicating that the weld connection design of the rack is adequate.

2.3.1.2 Revised Storage Rack Structural Analysis

The licensee performed additional analysis to verify that the modifications required to fit rack N3 in the pool did not adversely affect the original structural analysis conclusions. The licensee repeated all 20 simulations for this rack, using the modified 8 x 12 cell configuration instead of the original 8 x 13 configuration. The licensee demonstrated that the modified rack responded in a manner similar to the original design, and that adequate margin to design limits was preserved.

2.3.1.3 Storage Rack Analysis Conclusion

Based on (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME Code, and (3) the licensee's overall structural integrity conclusions supported by the SR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

2.3.2 Spent Fuel Storage Pool

The licensee indicated in its October 14, 1997 submittal that a comprehensive, detailed pool structure analysis was performed and submitted in the previous license amendment prepared for the reracking Campaign II. This analysis was reviewed and accepted by NRC in a license amendment issued on December 31, 1991. The licensee indicates that the factors of safety for bending moments and axial forces of the concrete walls and slab remain adequate due to the fact that the increased loading to the pool structure represented by the proposed 7 new high density storage racks fully filled with fuel assemblies for the reracking Campaigns III and IV remains below the loading considered in the previous analysis. In view of the NRC's previous acceptance and no changes in structural loading conditions in the previous pool structural analysis, the staff finds that the storage fuel pool structural design is acceptable.

2.3.3 Fuel Handling Accident Structural Effects

Two refueling accident cases were evaluated by the licensee: (1) drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of Accident Case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the fuel

assembly drop through the rack structure. The analysis results of Accident Drop Case (2) show that the maximum stress at the top of the rack is less than material yield stress. Thus, the functionality of the rack is not affected. The staff reviewed the licensee's analysis results and concurs with its findings. This is acceptable based on the licensee's structural integrity conclusions supported by the parametric studies.

2.3.4 Structural Integrity Conclusion

Based on the review and evaluation of the licensee's October 14, 1997 submittal, and additional information and analysis provided by the licensee on July 23, 1998, the staff concludes that the licensee's structural analysis and design of the spent fuel rack modules, and the SFP structures are adequate to withstand the effects of the applicable loads including the SSE. The analysis and design are in compliance with the current licensing basis set forth in the FSAR, and applicable provisions of the SRP and are, therefore, acceptable.

2.4 RADIOLOGICAL EFFECTS

2.4.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the modification of the FitzPatrick spent fuel racks with respect to occupational radiation exposure. As stated above, for this modification, the licensee plans to add seven new fuel rack modules to the SFP. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed for between 3 and 4 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. The Radiation Protection department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels (including potential exposure to hot particles), and dosimetry requirements. Each member of the project team will be required to attend an ALARA Pre-Plan meeting and each team member will be required to attend daily pre-job briefings on the scope of the work to be performed. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment including alarming dosimeters.

Since this license amendment does not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to utilize divers to remove any interferences that may impede the installation of the new spent fuel racks, the licensee will equip each diver with radiation detectors with remote, above surface readouts which will be continuously monitored by Radiation Protection personnel. Divers will also have access to an underwater survey meter, if needed. The divers will be in continuous communication with Radiation Protection personnel. The licensee will conduct

radiation surveys of the diving area prior to each diving operation and following the movement on any irradiated hardware. In order to minimize diver dose, the licensee will use visual barriers (such as air bubbles, ropes, or signs) as practical. In order to ensure that divers maintain a safe distance from irradiated sources, divers' movements may be restricted by the use of an umbilical. The diving area will be well-illuminated and the licensee will conduct continuous TV monitoring of the diver's location and work activities.

Any interferences or SFP hardware that must be removed from the SFP to permit rack installation will be decontaminated and removed from the SFP if dose levels permit. The licensee will monitor any items removed from the SFP for hot particles. Radioactive material removed from the SFP will be processed in accordance with station procedures and stored or shipped accordingly. The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase due to the expanded SFP storage capacity. However, the licensee will operate a continuous air monitor in the SFP area during the entire fuel racking operation to detect any potential airborne radioactivity. In addition, the plant effluent radiation monitoring system will monitor any gaseous releases.

The licensee will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Any remote tools utilizing hollow poles for handles will have holes drilled in the poles permitting water to fill the cavity, thereby eliminating the potential for radiation streaming while using these tools in the SFP. The licensee will wash/rinse any materials/components that come into contact with irradiated components prior to removal from the SFP and decon them to less than 100,000 dpm to reduce general contamination levels and to remove any hot particles.

The licensee plans to use an underwater vacuum to remove crud and debris from the bottom of the SFP during fuel rack installation to minimize any potential radiological effects of spalling from the fuel elements. This vacuum system will also be used to capture metal filings generated by any cutting performed in the SFP. The licensee will use the existing SFP filtration system during fuel rack installation to maintain water clarity in the SFP.

The storage of additional spent fuel assemblies in the SFP will not increase the dose rates on the refueling floor or in adjacent accessible areas to the SFP. The dose rate at the SFP surface due to gamma radiation from the fuel will be negligible. In order to ensure that area dose rates in adjacent accessible areas to the SFP do not exceed the maximum dose rate levels for which these areas are zoned, the licensee has administrative controls in place which specify that freshly discharged fuel (fuel which has been out of the core for less than a year) cannot be stored in designated storage cell locations adjacent to the SFP walls.

The licensee will install portable area radiation monitors at SFP rails adjacent to the work area for early detection of any floating hot particles and to monitor the work area. The average dose rates measured at the refueling bridge are 5-7 mrem/hr. Dose rates on the fuel pool level are primarily due to radionuclides in the pool water. The licensee does not expect these dose rates to increase due to the increased amount of spent fuel which will be stored in the SFP.

The staff finds these dose rates to be acceptable and in accordance with SFP dose rates at other plants.

On the basis of NRC staff review of the FitzPatrick proposal, the staff concludes that the FitzPatrick SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds the projected dose for the project of 3 to 4 person-rem to be in the range of doses for similar SFP modifications at other plants and is, therefore, acceptable.

2.4.2 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP purification system. The licensee does not expect the resin changeout frequency of the SFP purification system to be permanently increased as a result of the storage of additional spent fuel assemblies in the SFP. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove any radioactive crud and other debris before the new fuel rack modules are installed. Additional solid waste may be generated by the removal of any interferences or SFP hardware that may have to be removed from the SFP to permit rack installation. Overall, however, the licensee does not expect that the additional fuel storage made possible by the increased storage capacity will result in a significant change in the generation of solid radwaste.

2.4.3 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of a fuel handling accident to determine the thyroid and whole-body doses at the Exclusion Area Boundary (EAB), Low-Population Zone (LPZ), and Control Room. The proposed reracking of the FitzPatrick SFP will not affect any of the assumptions or inputs used in evaluating the dose consequences of the fuel handling accident.

The staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the licensee's doses. In performing these calculations, the staff used the assumptions of Regulatory Guide (RG) 1.25, Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors. For the fuel handling accident, the staff assumed that the cladding of 125 fuel rods would be ruptured if a fuel assembly were dropped during handling. The damaged fuel rods are assumed to contain freshly off-loaded fuel with a minimum of 24 hours of decay. The parameters that the staff utilized in its assessment are presented in Table 1.

The staff's calculations confirmed that the thyroid doses at the EAB, LPZ, and Control Room from a fuel handling accident meet the acceptance criteria and that the licensee's calculations are acceptable. The results of the staff's calculations are presented in Table 2. For a fuel handling accident, the staff calculated a dose of 1.08 rem thyroid at the EAB and 0.54 rem

thyroid at the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is contained in SRP Section 15.7.4 of NUREG-0800 and is 75 rem thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem). In calculating the dose to the control room operator from a fuel handling accident, the staff assumed that the control room was not manually isolated until 12 minutes into the accident. The staff also assumed that motor-operated valve MOV-108 in the control room air supply duct fails in the open position and a bypass damper downstream of this valve (DMPR-105) is not closed until 12 hours into the accident (as per accident description in the licensee's submittal). The staff calculated the resulting dose to the control room operator of 0.76 rem thyroid. The acceptance criterion for the control room operator is 30 rem thyroid (SRP Section 6.4 of NUREG-0800). Therefore, the staff finds the proposed reracking at FitzPatrick to be acceptable with respect to potential radiological consequences as a result of a hypothetical fuel handling accident.

2.5 CRITICALITY EVALUATION

The proposed TS provide limits for the infinite neutron multiplication factor, k_∞, for the fuel loaded in the racks, the effective neutron multiplication factor, k_{eff}, and on the spacing of fuel assemblies within the fuel racks. The spent fuel rack design includes a neutron absorber, boron, to limit the reactivity of the fuel stored in those racks. The boron for the FitzPatrick spent fuel racks is in the form of Boral, composed of aluminum and boron carbide. Boral as the boron containing material has been approved in over 30 previous NRC reviews. The Boral is fastened to the fuel cells and provides a high thermal neutron removal cross section, and has proven to be structurally sound in fuel pool applications.

The current, NRC-approved, FitzPatrick analysis approach and the Technical Specifications for the SFP and existing racks state that the reactivity status, k_{eff}, of the spent fuel pool shall be less than 0.95 at a 95% probability and confidence uncertainty level. This limit meets the NRC staff reactivity requirement. The specification further indicates that this k_{eff} value is satisfied if the maximum k_∞ of each of the stored fuel assemblies is no greater than 1.36. The present submittal does not propose to change this analytical approach or the spent pool criterion, which remains at 0.95 for both the old and new racks. The maximum k-infinity for the fuel assemblies has been reduced to 1.32. (k_∞ is calculated with an infinite array of specified, uncontrolled assemblies in a cold, 20 °C, reactor core configuration). The fuel assembly chosen for the k_∞ and corresponding pool analyses was the GE 12 fuel assembly configuration with a 4.6 weight percent (w/o) U-235 content. GE 12 was chosen because it has the highest reactivity for a given enrichment and gadolinium loading. The 4.6 w/o enrichment should encompass most future loadings, but this is not a requirement since the loading will have to meet the primary k_{eff} and k_∞ requirements.

The nuclear design and safety analysis was done by Holtec International. The analysis was performed using primarily the MCNP code, a 3-dimensional transport theory code developed by Los Alamos National laboratory, using continuous energy cross-sections and Monte Carlo random walk technique. Supplemental calculations, for verification, were done with NITAWL-KENO, a 3-dimensional code developed by Oak Ridge National Laboratory, also using Monte Carlo. The 238 group SCALE cross-sections were used. The CASM03 code

version 4.4, a transport theory code, was also used for some calculations of temperature and tolerance reactivity effects. These methodologies and cross sections are well known and have been accepted in past NRC reviews, including previous analyses by Holtec. The use of the two codes for primary analyses provides greater assurance for the analysis accuracy.

The methodologies and cross-sections have been benchmarked, by Holtec (and many other groups) against a number of relevant critical experiments simulating parameters related to storage racks. These benchmark calculations have been used to develop methodology bias and uncertainty factors to be added to the nominal k_{eff} calculations for the racks. Holtec has also determined the potential variation of rack and fuel parameters which are used in determining the k_{eff} of the rack and fuel system. These parameters include rack manufacturing tolerances, boron loading variations, Boral width tolerance variation, and cell lattice pitch variation. The variation of k_{eff} with these parameters (taken at a 95/95 probability/confidence level was determined. These (independent parameters.) were statistically combined with the methodology uncertainty to provide a Δk uncertainty, which was added to the base k_{eff} calculation. This treatment of the uncertainties is in conformance with NRC past recommendations and approvals. In addition rack calculations were done using conservative three-dimensional infinite arrays of cells and infinite fuel lengths.

Holtec has also investigated abnormal conditions that might be associated with the SFP. These include (1) pool water temperature effects (reference temperature was 20 °C, but a worse case temperature, 4 °C, was assumed for the investigation) (the moderator temperature reactivity coefficient is negative so that temperature increases or boiling reduce reactivity), (2) eccentric fuel positioning (the nominal analysis case with the fuel centered in the cell yields maximum reactivity), (3) dropped fuel assembly (no significant reactivity increase), and (4) rack lateral movement (no significant reactivity increase). These analyses have provided a satisfactory demonstration that reasonably possible abnormal conditions will not lead to a reactivity problem if the required k-infinity and k_{eff} limits are met.

For the new (unburned) fuel racks the TS retain the current approved TS k_{eff} limits of 0.90 if dry and 0.95 if fully flooded. There is a proposed maximum k-infinity limit of 1.31 for the fuel that can be in the racks. The introduction of a k-infinity limit is an acceptable improvement over the current new fuel rack TS which does not provide a k-infinity approach with such a specific criterion. Similar to the review for the SFP, this is an acceptable approach and specification.

There is no discussion in the FitzPatrick submittal of low density water moderation in connection with the new (unburned) fuel racks. Reactivity can be maximized in some storage configurations with low density moderation, such as that from fire fighting water sprays. This configuration needs to be addressed for new fuel racks unless it is shown that it will not occur. The possibility for occurrence has been previously addressed for FitzPatrick in the course of the NRC staff review for exemption from the requirements of 10 CFR 70.24 regarding criticality accidents requirements and criticality monitors (submittal dated April 24, 1998, and exemption authorized on June 24, 1998). The NRC staff concluded that "On the basis of the information provided, there is reasonable assurance that irradiated and unirradiated fuel will remain subcritical." This review and conclusion is acceptable for the present review of revised TS for the new fuel racks.

The NRC staff has reviewed the reports submitted by New York Power Authority, describing the addition of fuel racks to the SFP, the criticality analyses performed and methods used and the changes to the TS (for both the SFP and for the new fuel racks) resulting from the analyses. Based on this review, the staff concludes that appropriate documentation was submitted and that the proposed changes satisfy the staff positions and requirements in these areas. The criticality aspects of the spent fuel racks and the new (unburned) fuel racks are acceptable.

2.6 EDITORIAL CHANGES

The licensee proposes to renumber TS 5.1.A as 5.1.1, and to change the word "country" to "county." The word change is an editorial correction to the description of the site location. The licensee also proposes to renumber TS 5.1.B, 5.2.A, 5.2.B, 5.4.A, 5.4.B, and 5.4.C as 5.1.2, 5.2.1, 5.2.2, 5.4.1, 5.4.2, and 5.4.3, respectively. These changes are all editorial in nature, and do not affect any technical requirements associated with facility operation. Therefore, the proposed changes are acceptable.

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 (64 FR 61167) in the Federal Register on November 9, 1999. Based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

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

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Table 1

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

Parameters	
Power Level, Mwt	2586.4
Number of Fuel Rods Damaged	125
Total Number of Rods in Core	36,472
Shutdown Time, hours	24
Power Peaking Factor	1.5
Fission-Product Release Fractions (%)*	
Iodine (corrected for extended burnup)	12
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
SGTS Filter Efficiency (%)	90
Filter Efficiencies for Control Room (%)	
Elemental	90
Organic	90
Control Room Flow Rates (ft^3/min)	
Time = 0 to 12 min.	
Unfiltered air intake	15000
Time = 12 min. to 12 hrs	
Recirculation flow	0
Filtered air intake	1000
Unfiltered air intake	2100
Time = 12 hrs and later	
Filtered air intake	1000
Unfiltered air intake	300
Atmospheric Dispersion Factors, X/Q (sec/m^3)	
Exclusion Area Boundary (0-2 hours)**	5.18×10^{-5}
Low Population Zone (0-8 hours)**	2.6×10^{-5}
Control Room (0-8 hours)**	8.1×10^{-5}
Dose Conversion Factors per ICRP 30	

* Regulatory Guide 1.25
** Staff calculated

TABLE 2

THYROID DOSES FROM FUEL HANDLING ACCIDENT
AT FITZPATRICK (VALUES CALCULATED BY NRC STAFF)

	FUEL HANDLING ACCIDENT DOSE (REM)
EAB*	1.08
LPZ*	0.54
Control Room**	0.76

*Acceptance Criterion = 75 rem thyroid
**Acceptance Criterion = 30 rem thyroid