UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-59

THE POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 Introduction

By letter dated July 26, 1978, as supplemented[(a)], the Power Authority of the State of New York (the licensee or PASNY) proposed to change the spent fuel pool (SFP) storage design for James A. FitzPatrick Nuclear Power Plant (FitzPatrick) from the design which was reviewed and approved in the operating license review and described in the FSAR. The proposed change consists of increasing the total spent fuel storage capacity of the SFP from 760 fuel assemblies to 2244 fuel assemblies.

2.0 Discussion

The proposed spent fuel storage racks are to be made up of alternating, double-walled aluminum containers. These will be about 14 feet long and will have a square cross section with an inner dimension of 6.16 inches. The nominal pitch between fuel assemblies is 6.625 inches. The outer dimension of the square fuel assemblies that are to be stored in these racks is 5.12 inches. This results in an overall fuel region volume fraction of 0.60 in the nominal storage lattice cell. A Boral plate is to be seal welded in the cavity between the double walls. Thus, in this arrangement there will be only one Boral plate between adjacent fuel assemblies. In its submittal PASNY states that the minimum amount of boron-ten per unit area of Boral plate will be 0.0232 grams per square centimeter. This is equivalent to 1.4×10^{21} boron-ten atoms per square centimeter.

3.0 Description of the Proposal

The proposed SFP modification consists of replacing the existing fuel storage racks with new spent fuel racks to increase the storage capacity from 760 to 2244 fuel assemblies. The fuel assemblies are stored in anodized aluminum modules. The modules are interconnected in a group to minimize relative displacement and prevent impact. Each module is arranged in a 8 X 10, 8 X 8, or an 11 X 10 array. The fuel assemblies are inserted into cavities that are formed by a c[illegible]ter of cans that are arranged in a checker board

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(a) Supplemental letters dated May 15, June 22, September 25, October 10, and November 29, 1979, April 1, April 31, and October 31, 1980.

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pattern. The can provides separation and lateral restraint for each fuel assembly. Boral (B_4C) poison material is sealed in cavities within each can by welding. The cans are constrained by upper and lower castings that are bolted to plates along the perimeter to form a box structure. The lower casting vertically supports each fuel assembly. Each module is free-standing with no lateral restraints to the wall and is supported by four steel feet that transfer load to the pool floor. The lateral loads on the racks will be transferred by friction between the feet and the pool floor.

The new spent fuel racks will be installed on a phased basis to provide additional capacity as required during normal refueling outages. Installation has been sequenced to eliminate any interfacing between the existing racks and the new racks. During periods of phased installation, both groups of racks will be seismically supported. At no time will any object be moved over stored spent fuel in accomplishing these procedures.

3.1 Criticality Analyses

As stated in PASNY's July 26, 1978 submittal, the fuel pool criticality calculations are based on unirradiated BWR fuel assemblies with no burnable poison and a fuel loading of 14.8 grams of uranium-235 per axial centimeter of fuel assembly.

The Nuclear Associates International Corporation (NAI) performed the criticality analyses for PASNY. NAI made parametric calculations by using the CHEETAH-B computer program to obtain four-group cross sections for PDQ-7 diffusion theory calculations. The effective boron cross sections for the Boral plates were calculated with the CORC-Blade program. NAI stated that these programs have been extensively tested by using them to make benchmark experiment calculations and core physics calculations for several existing operating power reactors.

These computer programs were used to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice at 20°C with the minimum boron concentration in the Boral, i.e., 0.0232 grams of boron-ten per square centimeter. NAI then performed calculations to determine: (1) the highest neutron multiplication factor as a function of pool water temperature; (2) the effect of a possible reduction in the lattice pitch; and (3) the effect of eccentrically positioning fuel assemblies in the storage lattice. These calculations showed that when all of these effects are accounted for, the maximum effective neutron multiplication factor (K_{eff}) in the fuel pool will be less than 0.894. The accuracy of the diffusion theory method for this storage rack application was then checked by calculating the nominal reference case with the KENO-IV Monte Carlo program using 123 group cross sections from the GAM-THERMOS library, and it was found that the results of the diffusion theory method are accurate within one percent Δk.

Since it will be possible to inadventently place a fuel assembly between the outer periphery of a loaded rack and the walls of the fuel pool, NAI calculated the possible increase in the neutron multiplication factor for this event. In this calculation it was assumed that there was no Boral plate between the external assembly and the adjacent assembly in the rack. NAI found that the increase in the neutron multiplication factor will be less than 0.005. Thus, this event would increase the maximum possible k_{eff} to 0.889.

In order to have proof that the required amount of boron remains in the plates throughout the life of the racks, PASNY's proposal states that sealed Boral coupons will be provided for inservice surveillance.

3.1.1 Evaluation

The above described results compare favorably with the results of parametric calculations made with other methods for similar fuel pool storage lattices. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

In addition to the Quality Assurance Program which is described in the proposal, the NRC requires an onsite neutron attenuation test to verify with ninety five percent confidence that there are a sufficient number of the Boral plates in the racks so that the maximum k_{eff} will not be greater than 0.95.

We find that all factors that could affect the neutron multiplication factor in this pool have been conservatively accounted for and that the maximum neutron multiplication factor in this pool with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a SFP. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margins to preclude criticality in the fuel. Accordingly, there is a Technical Specification which limits the effective neutron multiplication factor in all SFPs to 0.95.

3.1.2 Conclusion

We find that when any number of the fuel assemblies, which PASNY described in these submittals, which have no more than 14.8 grams of uranium-235 per axial centimeter of fuel assembly, or equivalent, are loaded into the proposed racks, the k_{eff} in the fuel pool will be less than the 0.95 limit. We also find that in order to preclude the possibility of the k_{eff} in the fuel pool from exceeding this 0.95 limit without being detected, it is necessary, pending an NRC review, to prohibit the use of these high density storage racks for fuel assemblies that contain more than 14.8 grams of uranium-235, or equivalent, per axial centimeter of fuel assembly. On the basis of the information submitted, and the k_{eff} and fuel loading limits stated above we conclude that the health and safety of the public will not be endangered by the use of the proposed racks.

3.2 Spent Fuel Cooling

The licensed thermal power for FitzPatrick is 2436 MWth. PASNY plans to refuel this plant annually. In the annual cycle, about 140 of the 560 fuel assemblies in the core are replaced. To calculate the maximum heat loads in the spent fuel pool PASNY assumed a 150 hour time interval between reactor shutdown and the time when 140 fuel assemblies were transferred to the spent fuel pool and a 250 hour time interval between reactor shutdown and the time when 560 fuel assemblies were transferred to the spent fuel pool. For the power history prior to shutdown PASNY assumed an energy production of 27,558 MWD/MTU with a continuous energy density of 24 MW/MTU. With these assumptions PASNY used the method given in the NRC Standard Review Plan 9.2.5 to calculate the maximum possible heat loads for the modified SFP. For these cooling times and fuel burnups PASNY calculated the maximum heat load in the spent fuel pool to be about 10. X 10^6 BTU/hr for the final refueling which fills the pool, and 24. X 10^6 BTU/hr for a full core offload which fills the pool after twelve annual refuelings.

The spent fuel pool cooling system consists of two pumps and two heat exchangers. Each pump is designed to pump 525 gpm (2.63 X 10^5 pounds per hour). Both heat exchangers when fed by a single SFP cooling pump are designed to transfer 6.3 X 10^6 BTU/hr from 125° fuel pool water to 95°F Reactor Building Closed Loop Cooling Water which flows through the shell side of each heat exchanger at the rate of 467 gpm (2.34 X 10^5 pounds per hour). PASNY stated that 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 water temperature at or below 135°F.

Makeup water for the SFP is obtained from the seismic Category I Condensate Storage System, which has two 200,000 gallon storage tanks.

3.2.1 Evaluation

We find that PASNY's calculated peak heat loads for the modified pool with a storage capacity for 2244 fuel assemblies are conservative and acceptable. We also find that the maximum incremental heat load that will be added by increasing the number of spent fuel assemblies that are to be stored in this pool from 760 to 2244 will be 2.0 X 10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools.

We calculate that with both pumps operating, the spent fuel pool cooling system can maintain the fuel pool outlet water temperature below 125°F for a peak annual refueling heat load of 10. X 10^6 BTU/hr. We find that when the RHR system is aligned with the spent fuel pool cooling system, the combined system will have sufficient capacity to keep the SFP outlet water temperature below 135°F for a full core heat load of 24. X 10^6 BTU/hr.

Assuming an initial maximum average fuel pool water temperature of 125°F, the minimum time to achieve bulk boiling after any credible accident will be about nine (9) hours. This assumes no cooling of the SFP during that time interval. In order to preclude actual boiling, it is PASNY's intention to use a single RHR train to cool the SFP when an accident makes the normal SFP cooling system inoperable. During normal power operation the RHR system is unavailable for SFP cooling since both trains of this system must be available for the LPCI post accident ECCS function. In order to make the RHR system available, PASNY has instituted procedures which allow prompt reactor cooldown. Cold shutdown conditions are achieved in a timely manner (i.e., less than 9 hours) at which time the train of the RHR system can be lined up to the SFP. This will preclude pool boiling.

Although SFP boiling will not occur, to be conservative such an occurrence has been considered. Assuming the same 125°F SFP water temperature, bulk boiling will likewise occur within nine hours. Such boiling will result in a maximum evaporation rate of fifty gallons per minute. Within this time an equivalent makeup rate can be established from the condensate storage system. In addition, we also find that under bulk boiling conditions the fuel temperature will not exceed 350°F. In sum, we find that makeup capability will preclude fuel element uncovery and that the maximum temperature does not unacceptably effect fuel element integrity or surface corrosion.

3.2.2 Conclusion

We find that the present cooling capacity in the FitzPatrick SFP will be sufficient to handle the incremental heat load that will be added by the proposed modification. We also find that this incremental heat load will not alter the safety considerations of SFP cooling from that which we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed design.

3.3 Installation of Racks and Fuel Handling

About 429 out of the 760 storage spaces that are presently in the SFP are filled with spent fuel assemblies. Thus, about 45 percent of the pool will not have fuel assemblies in it at the time PASNY is proposing to change the racks. In this regard PASNY states that a portion of the south end of the pool has already been cleared of existing racks. PASNY also states that vacant racks in the south end of SFP will be cleared prior to installation of the new racks, i.e., additional racks will be removed. PASNY also states that during the installation of the new racks, administrative controls will be placed on the reactor building crane to insure that the racks cannot be lifted or carried over spent fuel.

3.3.1 Evaluation

Since about forty five percent of the pool will not have fuel assemblies in it when the racks are changed, PASNY should have no difficulty in keeping the racks that are being moved away from the spent fuel that is presently in the pool.

After the racks are installed in the pool, the fuel handling procedures in, and around, the pool will be the same as those procedures that were in effect prior to the proposed modifications.

3.3.2 Conclusion

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

3.4 Evaluations

3.4.1 Spent Fuel Handling Evaluation

The NRC staff has underway a generic review of load handling operations in the vicinity of SFPs to determine the likelihood of a heavy load impacting fuel in the pool and, if necessary, the radiological consequences of such an event. We have concluded that the likelihood of a heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFP are necessary while our review is underway

The consequences of fuel handling accidents in the SFP are not changed from those presented in the Safety Evaluation (SE) dated November 1972.

3.4.2 Structural and Mechanical Evaluation

The design and fabrication of the racks are in accordance with "Aluminum Construction Manual - Second Edition, Nov. 1971, Specifications of Aluminum Structures"; "Aluminum Standards and Data - Aluminum Association, 5th Edition, Jan. 197[illegible]", "Steel Construction Manual AISC (7th Edition), June 1973, American

Institute of Steel Construction"; and "ASME Boiler & Pressure Vessel Code, Section III, Subsection NA, Appendix I and XVII, 1974 Edition". The loads, load combinations and acceptance criteria used for the rack design are consistent with Sections 3.8.4.II.3 and 3.8.4.II.5 of the Standard Review Plan for steel structures. The materials and fabrication processes are essentially the same as those used at the Yankee Nuclear Power Station, which has performed satisfactorily for over 10 years.

The seismic design of the racks is based upon a nonlinear dynamic analysis using the ANSYS computer program that was developed by Swanson Analysis Systems, Inc. Seismic excitation along three orthogonal directions was used in the design. Floor acceleration time histories corresponding to SSE and OBE group acceleration levels of 0.15g and 0.08g were imposed. The analysis includes the effects of friction between the rack and floor, gaps between the fuel assembly and can, rack uplift, and fluid coupling due to the constrained water within the rack structure. No benefit was taken for the damping effect of water surrounding the rack. A low value of friction was used to maximize the predicted sliding displacement, while a high value of friction was used to maximize horizontal rocking displacement at the top of the rack. Under the most severe loading conditions the racks slide a maximum of 1.472 inches. A distance of 3.0 inches will be maintained between the rack and any rigid object within the pool. In addition, a distance of 6.05 inches will be maintained between any rack and the pool walls. Fuel assemblies were conservatively assumed to impact with the cans all at the same time. The integrity of the fuel cladding will be maintained under these conditions. Furthermore, rack to rack impact was considered although the racks are constrained to minimize relative motion. The worst case of two fully loaded racks impacting under a loading of highest sliding was evaluated assuming that all momentum is transferred from one rack to the other. The maximum rack-to-rack impact forces have been calculated to be 81,000 pounds for the SSE and 64,000 pounds for the OBE. The forces occur at the top grid only and are included in the stress analysis of this member. Since the upper fitting of the fuel assemblies is not attached to the top grid, these impact loads are not directly transmitted to the fuel assemblies.

All three components of earthquake have been conservatively considered in the rack design. As explained in the licensee's submittal, the time history analysis was done for only two components of earthquakes which were the maximum horizontal (X - direction and Y - direction). However, the forces computed from this planar time-history model were applied on the detail (3-D) model simultaneously in both the X-Y and Z-Y planes. These resultant loads were then combined by SRSS to obtain the overall loads. This method, in effectively considering all three components of earthquake, doubles up on the vertical (Y-direction loading).

The new racks have been analyzed to determine the effects of a dropped fuel assembly impacting at critical locations on the upper and lower castings. In all cases there is no change to the center to center spacing of the fuel and there is no dislocation of the Boral neutron absorbing material.

3.4.2 Conclusion

The licensee performed a review of the load carrying ability of the SFP structure and found that the existing structure is capable of supporting the increase in overall loading as a result of the proposed fuel pool modification. The steel liner and concrete floor slab were also evaluated for the effects of rack impact due to rocking to conform to the bearing stress and punching shear stress allowables of the American Institute of Steel Construction Specification for Steel Structures and the American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI 318-71). The temperature limits established in the FSAR for the pool remain the same and therefore the effects of temperature gradients on the pool structure will remain unchanged.

3.5 Occupational Radiation Exposure

We have reviewed the licensee's plan 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 this operation is estimated by the licensee to be about 6 man-rem. We consider this to be a reasonable estimate because it 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 was being performed. To ensure that the modification will be performed in a manner consistent with as low as is reasonably achievable (ALARA) occupational exposure, the licensee will remove unnecessary radioactive equipment and material from the fuel pool prior to the installation work, will store the spent fuel at the opposite end of the pool from where the installation work is being performed, and will pre-plan procedures necessary for the removal of the old racks and installation of the new ones. The existing low density racks will be decontaminated upon removal from the pool, packaged and shipped intact to a disposal site as low concentration radioactive waste.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel area from radionuclide concentrations in the SFP water and deposited on the SFP walls. 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. The occupational radiation exposure resulting from the additional spent fuel in the pool represents a negligible burden. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification should add less than one percent to the total annual occupational radition exposure burden at this facility. The small increase in additional exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

3.6 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluation dated November 1972. There will be no change in the waste treatment system or in the conclusion given in Section 8.2 of the evaluation of this system because of the proposed modification.

3.7 Materials

The purity of SFP water is maintained by a combination of filtering and ion exchange process to assure compatibility with the aluminum spent fuel racks. Significant corrosion of the rack structure or nuclear fuel components is highly unlikely to occur. The consequences of a Boral storage cavity weld leak have been evaluated and found to be negligible. However, a vacuum and pressure test is performed to assure the integrity of these welds.

3.7.1 Evaluation

The criteria used in the analyses, design and construction of the new spent fuel racks to account for anticipated loadings and postulated conditions that may be imposed upon the structure during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff. The use of these criteria provide reasonable assurance that the new fuel pool structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

3.7.2 Summary

Our evaluation supports the conclusion that the proposed modification to the FitzPatrick SFP is acceptable because:

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

(2) The potential consequences of the postulated design basis accident for the SFP, i.e., the rupture of the fuel pins in the equivalent of one fuel assembly and the subsequent release of the radioactive inventory within the gap, are acceptable

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

4.0 Technical Specification

As indicated in the criticality analysis of this Safety Evaluation and in the licensee's referenced submittals the maximum average Uranium-235 content is specified in Technical Specification 5.5 to be 3.3 w/o. Therefore, fuel assemblies that are bound by the fuel assembly designs described in the licensee's referenced submittals may be stored in the spent fuel pool. This will result in satisfying the facility design criteria of k_{eff} (dry) <0.90 and (flooded) <0.95.

As indicated in the spent fuel pool cooling analyses of this Safety Evaluation, the RHR system may be used to augment the spent fuel pool cooling system. Since such RHR usage would make the LPCI system unavailable, cross-tieing the RHR and SFP cooling system is allowable only during plant shutdowns. Therefore, in accordance with Technical Specification 3.5 the RHR system may be used for SFP cooling when reactor coolant temperature is below 212°F.

5.0 Conclusion

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.

Dated: June 18, 1981