Docket File

Docket No. 50-333

Distribution
Docket
ORB #2
Local PDR
NRC PDR
DEisenhut
RPurple
TNovak
RTedesco
GLainas
PPolk
TIppolito
SNorris
OELD
IE (5)
TBarnhardt (4)
BScharf (10)
JWetmore
ACRS (16)
OPA (CMiles)
RDiggs
HDenton
JHeltemes, AEOD
NSIC
TERA
ALee
KWichman
WHazelton
JDonohew
MWohl
ELantz

RECEIVED JUN 24 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. George T. Berry
President and Chief Operating
Officer
Power Authority of the State of
New York
10 Columbus Circle
New York, New York 10019

June 18, 1981

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 55 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Generating Plant. The amendment is in response to your application dated July 26, 1978, as supplemented by letters dated May 15, June 22, September 25, October 10, and November 29, 1979, April 1, April 31, and October 31, 1980.

This amendment will allow an increase in the spent fuel storage capability up to a maximum of 2244 fuel assemblies by use of high density spent fuel racks. Portions of your proposed Technical Specifications have been modified. These changes have been discussed and agreed to by members of your staff.

Copies of the Safety Evaluation, Environmental Impact Appraisal and Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

cp

Enclosures:
1. Amendment No. 55 to DPR-59
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice and Negative Declaration

cc w/enclosures:
See next page

| | | | | | | |
|---|---|---|---|---|---|---|
| | | | ORB #2 | ORB #2 | AD:OR | |
| | | | SNorris | PPolk:mjf | TNovak | ELantz |
| | | | 6/10/81 | 6/4/81 | 6/15/81 | 6/11/81 |

amdt + notice only

8106260386

| OFFICE | | | | | | OELD | ORB #2 |
|---|---|---|---|---|---|---|---|
| SURNAME | ALee | KWichman | WHazelton | TDonohew | MWohl | OELD | TIppolito |
| DATE | 6/11/81 | 6/11/81 | 6/11/81 | 6/11/81 | /81 | 6/16/81 | 6/12/81 |

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

June 18, 1981

Docket No. 50-333

Mr. George T. Berry
President and Chief Operating
Officer
Power Authority of the State of
New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

The Commission has issued the enclosed Amendment No. 55 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Generating Plant. The amendment is in response to your application dated July 26, 1978, as supplemented by letters dated May 15, June 22, September 25, October 10, and November 29, 1979, April 1, April 31, and October 31, 1980.

This amendment will allow an increase in the spent fuel storage capability up to a maximum of 2244 fuel assemblies by use of high density spent fuel racks. Portions of your proposed Technical Specifications have been modified. These changes have been discussed and agreed to by members of your staff.

Copies of the Safety Evaluation, Environmental Impact Appraisal and Notice of Issuance and Negative Declaration are also enclosed.

Sincerely,

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 55 to DPR-59
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice and Negative Declaration

cc w/enclosures:
See next page

Mr. George T. Berry
Power Authority of the State
of New York

cc:

Mr. Charles M. Pratt
Assistant General Counsel
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

U. S. Environmental Protection
Agency
Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

Mr. Raymond J. Pasternak
Resident Manager
James A. FitzPatrick Nuclear
Power Plant
P. O. Box 41
Lycoming, New York 13093

Director, Technical Development
Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

George M. Wilverding
Manager - Nuclear Licensing
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Mr. Robert P. Jones, Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

State University College at Oswego
Penfield Library - Documents
Oswego, New York 13126

Mr. J. Phillip Bayne
Senior Vice President -
Nuclear Generation
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Resident Inspector
c/o U.S. NRC
P. O. Box 136
Lycoming, New York 13093

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
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 July 26, 1978, as supplemented by filings dated May 15, June 22, September 25, October 10, and November 29, 1979, April 1, April 31 and October 31, 1980, 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.

8106260389

2. Accordingly, Facility Operating License No. DPR-59 is amended by changes to the Technical Specifications as indicated in the attachment, 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. 55 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 18, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 55

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A Technical Specifications as follows:

| Delete | Replace |
| --- | --- |
| 115 | 115 |
| 245 | 245 |
| 246 | 246 |

b. From the time that the LPCI mode is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the LPCI mode is made operable earlier provided that during these 7 days all active components of both Core Spray Systems, the containment spray subsystem (including two RHR pumps) and the emergency diesel generators shall be operable.

c. When the reactor water temperature is greater than 212°F the motor operator for the RHR cross-tie valve (MOV20) shall be maintained disconnected from its electric power source. It shall be maintained chain-locked in the closed position. The manually operated gate valve (10-RHR-09) in the cross-tie line, in series with the motor operated valve, shall be maintained locked in the closed position.

4. a. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

b. The RHR System shall not supply cooling to the spent fuel pool when the reactor coolant temperature is above 212°.

b. When it is determined that the LPCI mode is inoperable, both Core Spray Systems, the containment spray subsystem, and the emergency diesel generators shall be demonstrated to be operable immediately and daily thereafter.

c. The power source disconnect and chain lock to motor operated RHR cross-tie valve, and lock on manually operated gate valve shall be inspected once each operating cycle to verify that both valves are closed and locked.

## 5.0 DESIGN FEATURES

### 5.1 SITE

A. The James A. FitzPatrick Nuclear Power Plant is located on the PASNY portion of the Nine Mile Point site, approximately 3,000 ft. east of the Nine Mile Point Nuclear Station. The NMP-JAF site is on Lake Ontario in Oswego County, New York, approximately 7 miles northeast of Oswego. The plant is located at coordinates north 4,819, 545.012 m, east 386,968.945 m, on the Universal Transverse Mercator System.

B. The nearest point on the property line from the reactor building and any points of potential gaseous effluents, with the exception of the lake shoreline, is located at the northeast corner of the property. This distance is approximately 3,200 ft. and is the radius of the exclusion areas as defined in 10 CFR 100.3.

### 5.2 REACTOR

A. The reactor core consists of not more than 560 fuel assemblies. For the current cycle four fuel types are present in the core: 7 x 7, 8 x 8, 8 x 8R and P8 x 8R. These fuel types are described in Section 3.2 of the FSAR and NEDO-24011. The 7 x 7 fuel has 49 fuel rods, the 8 x 8 fuel has 63 fuel rods and 1 water rod, and the 8 x 8R and P8 x 8R fuel have 62 fuel rods and 2 water rods.

B. The reactor core contains 137 cruciform-shaped control rods as described in Section 3.4 of the FSAR.

### 5.3 REACTOR PRESSURE VESSEL

The reactor pressure vessel is as described in Table 4.2-1 and 4.2-2 of the FSAR. The applicable design codes are described in Section 4.2 of the FSAR.

### 5.4 CONTAINMENT

A. The principal design parameters and characteristics for the primary containment are given in Table 5.2-1 of the FSAR.

B. The secondary containment is as described in Section 5.3 and the applicable codes are as described in Section 12.4 of the FSAR.

C. Penetrations of the primary containment and piping passing through such penetrations are designed in accordance with standards set forth in Section 5.2 of the FSAR.

### 5.5 FUEL STORAGE

A. The new fuel storage facility design criteria are to maintain a $K_{eff}$ dry <0.90 and flooded <0.95. Compliance shall be verified prior to introduction of any new fuel design to this facility.

5.5 (cont'd)

B. The spent fuel storage pool is designed to maintain $K_{eff}$ <0.95 under all conditions as described in the Authority's application for spent fuel storage modification transmitted to NRC July 26, 1978. In order to assure that the criteria is met, new fuel average enrichment will be limited to $\leq$ 3.3 w/o U-235. The number of fuel assemblies stored in the spent fuel pool shall not exceed 2244.

## 5.6 SEISMIC DESIGN

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

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

# 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 $1.4 \times 10^{21}$ 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 cluster of cans that are arranged in a checker board

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

8106260392

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 $\Delta k$.

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 135°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. 1976"; "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

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

# ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO. 55 TO DPR-59

## POWER AUTHORITY OF THE STATE OF NEW YORK

## JAMES A. FITZPATRICK NUCLEAR POWER PLANT

## DOCKET NO. 50-333

## 1.0 Description of Proposed Action

In their submittal of July 26, 1978, as supplemented[a], the Power Authority of the State of New York (the licensee or PASNY) proposed to increase the total storage capacity of the spent fuel pool (SFP) at James A. FitzPatrick Nuclear Power Plant (JAFNPP) from 760 to 2244 fuel assemblies.

## 2.0 Need for Increased Storage Capacity

The JAFNPP SFP was originally designed with the storage capacity of 760 fuel assemblies (1 1/2 cores). The first refueling of JAFNPP occurred in the spring of 1977, at which time 140 fuel assemblies (1/4 of the reactor core) were replaced and stored in the SFP. The JAFNPP refuels on an annual basis. Therefore, at this rate, 140 assemblies per year are discharged from the reactor to the SFP. At this time, there have been 3 refuelings at JAFNPP and there are 429 spent fuel assemblies in the SFP. Fuel core offload capability is not presently possible and after two additional refuelings the SFP capacity will not exist to allow further refuelings.

Spent fuel is not currently being processed on a commercial basis in the United States and storage capacity away from reactor sites is available only on an emergency basis as is discussed in Sections 6.1 and 6.2 of this appraisal.

Based on the above information, there is clearly a need for additional onsite SFP storage capacity to assure continued operation of JAFNPP. The expansion of the SFP capacity to 2244 assemblies would provide capacity through the 1992 refueling outage.

## 3.0 The Facility

The JAFNPP is described in Section 3.0 of the Final Environmental Statement (FES), issued by the Commission in March 1973. The unit is a Boiling Water Reactor (BWR) which produces 2436 megawatts thermal (MWt) and has a gross electrical output of 821 megawatts (MWe). Pertinent descriptions of principal features of the plant as it currently exists are summarized below to aid the reader in following the evaluations in subsequent sections of this appraisal.

3.1 Fuel Inventory

The JAFNPP reactor contains 560 fuel assemblies. The fuel assemblies are arranged in an 8 by 8 array. The weight of the fuel, as $UO_2$, is approximately 265,000 pounds. About one forth of the assemblies are removed from the reactor and replaced with new fuel each year. Present scheduling is for the refueling outage to be in the fall of each year.

The proposed modification of the SFP would not change the quantity of uranium fuel used in the reactor over the anticipated operating life of the facility and would not change the rate at which spent fuel is generated by the facility. The added storage capacity would increase the number of spent fuel assemblies that could be stored in the SFP and the length of time that some of the fuel assemblies could be stored in the pool.

3.2 Purpose of the SFP

Spent fuel assemblies are intensely radioactive due to their fresh fission produce content when initially removed from the core and they have a high thermal output. The SFP was designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. The major portion of decay occurs in the first 150 days following removal from the reactor core. After this period, the spent fuel assemblies may be withdrawn and placed in heavily shielded casks for shipment. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling.

3.3 Spent Fuel Pool Cooling and Cleanup Systems

The SFP cooling and cleanup system consists of two skimmer surge tanks, two cleanup recirculating pumps, two heat exchangers, two filter demineralizers and the required piping, valves and instrumentation. The pumps draw water from the skimmer tanks and discharge it through the heat exchangers. The water is then passed through the filter-demineralizers or returned directly to the SFP. One filter-demineralizer is used normally but both are used during refueling operations. The Reactor Water Cleanup System filter demineralizers may also be used during refueling to reduce the burden on the SFP units.

Because it is expected that only a small increase in radioactivity released to the pool water as a result of the proposed modification as discussed in Section 4.1, we conclude that the SFP purification system will keep concentrations of radioactivity in the pool to levels which have existed prior to the modification.

3.4 Radioactive Wastes

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in Section 3.0 of Final Environmental

Statement (FES) dated March 1973. There will be no change in the waste treatment systems described in Section 3.0 of the FES because of the proposed modification.

## 4.0 Environmental Impacts of the Proposed Action

### 4.1 Land Use

The external dimensions of the SFP will not change because of the proposed expansion of its storage capacity; therefore, no additional commitment of land is required. The SFP is intended to store spent fuel assemblies under water for a period of time to allow shorter-lived radioactive isotopes to decay and to reduce their thermal heat output. This type of use will remain unchanged by the modification by the additional storage capacity would provide for an additional sixteen normal refuelings. Thus, the proposed modification would result in more efficient use of the land already designed for spent fuel storage.

### 4.2 Water Use

There will be no significant change in plant water consumption or use as a result of the proposed modifications. As discussed subsequently, storing additional spent fuel in the SFP will slightly increase the heat load on the SFP cooling system. This heat is transferred in turn to the component cooling water system and to the service water system. The modifications will not change the flow rate within these cooling systems. The temperature of the SFP water during normal refueling operations with only one SFP cooling pump running is expected to remain below 125°F, as compared to the 120°F used as the design basis in the FSAR. Therefore, the rate of evaporation and thus the need for makeup water will not be significantly changed by the proposed modifications.

### 4.3 Nonradiological Effluents

There will be no change in the chemical or biocidal effluents from the plant as a result of the proposed modification.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be additional discharge of heat to Lake Ontario. Storing spent fuel in the SFP for a longer period of time will not add significantly more heat to the SFP water. The SFP heat exchangers are cooled by the reactor building closed loop cooling water system which in turn is cooled by the plant cooling water system. The maximum heat load resulting from the SFP modification is $24X10^6$ BTU/hr. The small additional heat load from the SFP cooling system will be negligible since the additional storage capacity will not result in more spent fuel which has recently been placed in the SFP. Rather, it will result in more spent fuel stored for many years. The effect is negligible for older fuel since most thermal decay from the spent fuel occurs within one year; i.e., less than a refueling cycle.

## 4.4 Radiological

### 4.4.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is the oldest fuel which has not been shipped from the plant. This fuel should have decayed about five years. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the spent fuel pool water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the spent fuel pool during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the spent fuel pool cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800 F. A few weeks after refueling, the spent fuel cools in the spent fuel pool so that the fuel clad temperature is relatively cool, approximately 180 F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap.

4.4.2 Effect of Fuel Failure on the SFP

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP. During and after refueling, the SFP cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the SFP so that fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months.

Based on the operational reports submitted by the licensee and discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the Morris Operation (MO) (formerly Midwest Recovery Plant) at Morris, Illinois, or at the Nuclear

Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

Experience indicates that there is little radionuclide leakage from Zircaloy-clad spent fuel stored in pools for over a decade. Operators at several reactors have discharged, stored, and/or shipped relatively large numbers of Zircalogy-clad fuel elements which developed defects during reactor exposure, e.g., Ginna, Oyster Creek, Nine Mile Point, and Dresden Units Nos. 1 and 2. Based on the operational reports submitted by licensees and discussions with the operators, there has not been any significant leakage of fission products from spent reactor fuel stored in the MO pool or the NFS pool. Several hundred Zircaloy-clad assemblies which developed one or more defects in-reactor are stored in the MO pool without need for isolation in special cans. Detailed analysis of the radioactivity in the pool water indicates that the defects are not continuing to release significant quantities of radioactivity.

A Battelle Northwest Laboratory (BNL) report, "Behavior of Spent Nuclear Fuel in Water Pool Storage" (BNWL-2256 dated September 1977), states that radioactivity concentrations may approach a value up to 0.5 μCi/ml during fuel discharge in the SFP. After the refueling, the SFP ion exchange and filtration units will reduce and maintain the pool water in the range of $10^{-3}$ to $10^{-4}$ μCi/ml.

In handling defective fuel, the BNL study found that the vast majority of failed fuel does not require special handling and is stored in the same manner as intact fuel. Two aspects of the defective fuel account for its favorable storage characteristics. First, when a fuel rod perforates in-reactor, the radioactive gas inventory is released to the reactor primary coolant. Therefore, upon discharge, little additional gas release occurs. Only if the failure occurs by mechanical damage in the basin are radioactive gases released in detectable amounts, and this type of damage is extremely rare. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels. The second favorable aspect is the inert character of the uranium oxide pellets in contact with water. This has been determined in laboratory studies and also by casual observation of pellet behavior when broken rods are stored in pools.

4.4.3 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defected fuel. However, we have conservatively estimated that an additional 99 curies per year of Krypton-85 may be released from the SFP when the modified pool is completely filled. This increase would result in an additional total body dose of less than 0.001 mrem/year to an individual at the site boundary. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.005 man-rem/year. This is small compared to the fluctuations in the annual dose this population would receive from natural background radiation. This exposure represents an increase of less than 0.1% of the exposure from the plant evaluated in the FES. Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies in the pool may increase the bulk water temperature during normal refuelings above the 125°F used in the design analysis. The temperature may reach 125°F during the third refueling for less than one day and may exceed 125°F during future refuelings. During the sixteenth refueling, it may be above 125°F for about 14 days. The maximum temperature will be less than 135°F. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the SFP. Therefore, even if there were a slightly higher evaporation rate from the SFP, the increase in tritium and iodine released from the plant as a result of the increase in stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total release of gaseous radioactivity from the plant including the SFP.

### 4.4.4 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the filter-demineralizer and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pool and decreases as the pool water is processed through the filter-demineralizer. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste due to the modification, as a conservative estimate, we have assumed that the amount of solid radwaste may be increased by 50 cubic feet a year from the filter-demineralizer (an additional 10 resin bed/year). The amount of solid waste shipped from the site in 1980 was 28,000 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification systems by about 50 cubic feet per year, the increase in total waste volume shipped would be less than 0.17% and would not have any significant environmental impact.

The present spent fuel racks to be removed from the SFP because of the proposed modification are contaminated and will be disposed of as low level solid waste. The licensee has estimated that less than 7,200 cubic feet of solid waste will be removed from the plant because of the proposed modification. Therefore, the total waste shipped from the plant should be increased by less than 1% per year when averaged over the lifetime of the plant. This will not have a significant environmental impact.

4.4.5 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. The amount of radioactivity on the SFP filter-demineralizer might slightly increase due to the additional spent fuel in the pool, but this increase of radioactivity should not be released in liquid effluents from the plant. The plant radiological effluent technical specifications, which are not being changed by this action, restrict the total releases of liquid radioactivity from the plant.

The filter-demineralizer resins are periodically flushed with water to the waste sludge tank. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system.

Leakage from the SFP would be collected in the reactor building floor drain sumps through the pool leak detection system. The leakage would then be transferred to the liquid radwaste system and processed by the system before any water is discharged from the plant. There have not been signs of leakage from the pool.

4.4.6 Occupational Radiation Exposures

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire 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 dose rate measurements and occupancy factors for individuals performing a specific job during the modification. This operation is expected to be a small fraction of the total man-rem burden from occupational exposure.

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 occupancy times and dose rates in the SFP area. The spent fuel assemblies themselves will contribute a negligible amount of dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

4.4.7 Impacts of Other Pool Modifications

As discussed above, the additional environmental impacts in the vicinity of FitzPatrick resulting from the proposed modification are very small fractions (less than 1%) of the impacts evaluated in the FitzPatrick FES. These additional impacts are too small to be considered anything but local in character.

James A. FitzPatrick Nuclear Power Plant is located on the same site as Nine Mile Point Nuclear Station Unit 1 (NMP-1). By letter dated March 22, 1978, Niagara Mohawk Power Corporation proposed increasing the spent fuel storage capacity at NMP-1. Operation of NMP-1 was evaluated in the NMP-1 Final Environmental Statement dated January 1974.

The impact of any environmental significance at FitzPatrick from the proposed SFP modification at NMP-1 is the additional gaseous effluent from the NMP-1 SFP modification. We have conservatively estimated an additional 12 curies per year of Krypton 85 may be released from NMP-1 when its modified pool is completely filled. This additional Krypton 85 would result in an additional total body dose, that might be received by an individual near FitzPatrick or by the estimated population within a 50 mile radius, of less than 0.0002 mrem/year and 0.0005 man-rem/year, respectively.

Summing the additional exposures resulting from the SFP modifications at both NMP-1 and FitzPatrick shows the additional total body dose that might be received by an individual and by the estimated population out to 50 miles is less than .0012 mrem/yr and 0.0055 man-rem/yr, respectively. These summed exposures are small compared to the fluctuations in the annual dose this population receives from natural background radiation and represents an increase of less than 0.1% of the combined exposures evaluated in the FitzPatrick FES and the NMP-1 FES. These estimates are not significant.

Based on the above, we conclude that an SFP modification at any other facility should not significantly contribute to the environmental impact of FitzPatrick and that the FitzPatrick SFP modification should not contribute significantly to the environmental impact of any other facility.

4.4.8 Impacts on the Community

The new storage racks were fabricated offsite and shipped to the JAFNPP, where they are stored. Only a few truck or rail shipments would be involved in shipment of these racks and disposal of the present ones. The impacts of dismantling the present racks and installing the new ones will be limited to those normally associated with metal working activities. No significant impact on the community is expected to result from the fuel rack conversion or subsequent operation with increased storage of spent fuel in the SFP.

4.5 Evaluation of Radiological Impact

As discussed above, the proposed modification does not significantly change the radiological impact evaluated in the FES.

5.0 Environmental Impact of Postulated Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the FES for JAFNPP dated March 1973.

Additionally, 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. Because the JAFNPP has committed to prohibit the movement of heavy loads over the fuel assemblies in the SFP, 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.

6.0 Alternatives

The staff has considered the following alternatives to the proposed expansion of the SFP storage capacity at JAFNPP: (1) reprocessing the spent fuel; (2) shipment of spent fuel to a separate fuel storage facility; (3) shipment of spent fuel to another reactor site; (4) wet or dry storage onsite; (5) reduced plant operation; and (6) shutdown of facility. These alternatives are discussed below.

6.1 Reprocessing of Spent Fuel

As discussed earlier, none of the three commercial reprocessing facilities in the U.S. is currently operating. The General Electric Company's Midwest Fuel Recovery Plant at Morris, Illinois is in a decommissioned condition. On September 22, 1976, Nuclear Fuel Services, Inc. (NFS) informed the Nuclear Regulatory Commission that they were "withdrawing from the nuclear fuel reprocessing business". The NFS facility is on land owned by the State of New York. The lease to NFS expired in 1980. The Allied-General Nuclear Services (AGNS) reprocessing plant at Barnwell, South Carolina, received a construction permit on December 18, 1970. In October 1974, AGNS applied for an operating license for the reprocessing facility; construction of the reprocessing facility is essentially complete but no operating license has been granted. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU of spent fuel in the onsite storage pool, on which construction has also been completed but hearings with respect to this application have not been held and no license has been granted.

In 1976, Exxon Nuclear Company, Inc. submitted an application for a proposed Nuclear Fuel Recovery and Recycling Center (NFRRC) to be located at Oak Ridge, Tennessee. The plant would include a storage pool that could store up to 7,000 MTU in spent fuel. However, licensing review of this application was discontinued in 1977 as discussed below.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the U. S. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U. S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling".

On December 23, 1977, the NRC terminated the fuel cycle licensing actions involving mixed oxide fuel (GESMO) (Docket No. RM-50-5), the AGNS' Barnwell Nuclear Fuel Plant Separation Facility, Uranium Hexafluoride Facility and Plutonium Product Facility (Dockets Nos. 50-332, 70-1327 and 70-1821), the Exxon Nuclear Company, Inc. NFRRC (Docket No. 70-1432), and the NFS West Valley Reprocessing Plant (Docket No. 50-201). The Commission also announced that it would not at this time consider any other applications for commercial facilities for reprocessing spent fuel, fabricating mixed-oxide fuel, and related functions.

Although there are indications that the present Administrations policies on reprocessing and storage of spent fuel differ substantially from those of the previous Administration the staff considers that shipment of spent fuel to such facilities for reprocessing is not a feasible alternative to the proposed expansion of the JAFNPP SFP storage capacity, especially when considered in the relevant time frame, i.e., in the years 1981-1984 when the expanded storage will be needed. Even given a change in the governments reprocessing policy, the GESMO proceedings must be reopened and concluded, the licensing of the facilities must take place, the facilities must be constructed and brought on line before any fuel could be reprocessed. These things would likely require that the spent fuel be stored somewhere for up to another ten years.

6.2 Independent Spent Fuel Storage Facility

An alternative to expansion of onsite storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1,000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. The fuel storage pools at MO and NFS are functioning as smaller ISFSIs although this was not the original design intent. The license for the General Electric (GE) facility was amended on December 3, 1975 to increase the storage capacity to about 750 MTU; and, as of March 1, 1981, 316 MTU was stored in the pool in the form of 1220 spent fuel assemblies. An application for an 1100 MTU capacity addition is pending. However, by a motion dated November 8, 1977 GE requested the Atomic Safety and Licensing Board to suspend indefinitely further proceedings on this application. This motion was granted.

The staff has discussed the status of storage space at MO with GE personnel. We were informed that GE is primarily operating the MO facility to store either fuel owned by GE (which had been leased to utilities on an energy basis), or fuel which GE has previously contracted to reprocess. We were also informed that the present GE policy is not to accept spent fuel for storage except fuel for which GE has a previous commitment. There is no such commitment for JAFNPP spent fuel. Storage of the JAFNPP spent fuel at the existing reprocessing facilities is not a viable alternative to the expansion of the JAFNPP spent fuel pools.

The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool at West Valley. Although the storage pool is not full, NFS has indicated that it is not accepting additional spent fuel, even from the reactor facilities with which is has reprocessing contracts.

If the receiving and storage station at Barnwell is eventually licensed to accept spent fuel, as discussed in Section 6.1, it would be functioning as an ISFSI until the reprocessing facilities there are licensed to operate. The pool has unused space for about 400 MTU, but AGNS has indicated that it does not wish to operate the storage facility without reprocessing.

With respect to construction of new ISFSIs, on October 6, 1978 the NRC proposed a new Part 72 of its regulations specifying procedures and requirements for the issuance of relevant licenses, along with requirements for the siting, design, operation and record keeping activities of the facilities (43 FR 46309). In March 1981, the NRC issued for comment a draft regulatory guide and value/impact statement on preparation of a license application for an ISFSI. The staff has estimated that at least five years would be required for completion of an ISFSI. This estimate assumes one year for preliminary design; one year for preparation of the license application, environmental report, and licensing review for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for additional independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merrill Lynch, Pierce, Fenner and Smith, Inc, issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975 (ANS Transactions, 1975 Winter Meeting, Vol. 22, TANSAO 22-1-836, 1975). In 1974, E. R. Johnson Associates estimated the construction cost would be equivalent to approximately $9,000 per spent fuel assembly.

Several licensees have evaluated construction of an ISFSI and have provided cost estimates. In 1975, Connecticut Yankee, for example, estimated that an independent facility with a storage capacity of 1,000 MTU (BWR and/or PWR assemblies) would cost approximately $54 million and take about five years to put into operation. The Commonwealth Edison Company estimated the construction cost of an ISFSI in 1975 at about $10,000 per fuel assembly. To this would be added the costs for maintenance, operation, safeguards, security, interest in investment, overhead, transportation and other costs. These costs are significantly larger than the estimated cost of the increased storage capacity which will be obtained by expanding the present reactor pools (estimated by the licensee to be $6,400/assembly).

For the long term, DOE is modifying its program for nuclear waste management to include design and evaluation of a long term repository to provide Government storage of unreprocessed spent fuel rods in a retrievable condition. It is estimated that the long term storage facility will start accepting commercial spent fuel in the time frame of 1995 to 2000. The criteria for acceptance is that the spent fuel must have decayed a minimum of ten years so it can be stored in a dry condition without need for forced air circulation.

DOE has recently revised its policy with respect to the provision by DOE of interim fuel storage facilities. DOE has announced a decision to discontinue its efforts to provide Federal Government-owned or controlled away-from-reactor short term storage facilities. DOE intends to redirect its efforts to support the development of alternative means to be employed by utilities to further increase spent fuel storage capabilities. This leaves the task of developing interim storage capacity to private industry. Development of such capacity, if it is in the form of independent spent fuel storage installations, would most likely occur in conjunction with development of a reprocessing facility as discussed above and in Section 6.1. Since, as DOE has recently acknowledged, it may take some time for the reprocessing facilities, we conclude that an independent spent fuel storage installation will not be a feasible alternative to meet the licensee's needs within the time frame of interest, 1981-1984.

The staff does not regard the alternative of storing spent fuel at MO or Barnwell as offering a significant environmental advantage over construction and use of an expanded storage facility at JAFNPP. The availability of this alternative is speculative and it also would be considerably more expensive. Furthermore, constructing a new ISFSI would clearly have a greater environmental impact than the proposed action. It would require additional land and considerable equipment and structures, whereas installing new racks at JAFNPP requires only the small amount of material necessary to construct the racks and minor personnel exposure during installation.

## 6.3 Storage at Another Reactor Site

Niagara Mohawk Power Corporation owns and operates the Nine Mile Point (Unit 1) Nuclear Plant. The Nine Mile facility is also a boiling water reactor. However, the fuel handling and storage equipment for fuel assemblies from the two plants are not compatible. Niagara Mohawk is also confronted with the similar problem of spent fuel storage capacity. The licensee cannot assuredly rely on other power facilities to provide additional storage capability except on a short term emergency basis. If space were available in another reactor facility, the costs would probably be comparable to the cost of storage at a commercial storage facility.

6.4 Comparison of Alternatives

In Section 4 of this environmental impact appraisal the incremental environmental impacts of the proposed expansion of the SFP storage capacity were evaluated and were found to be insignificant. Therefore, none of the alternatives to this action offers a significant environmental advantage. Furthermore, alternatives (1), reprocessing, and (2), storage at an independent spent fuel storage facility, are not presently available to the licensee and are not likely to become available in time to meet the licensee's need. Alternative (3), shipment to another reactor site, would be a short term emergency solution but would eventually involve shipment to another temporary storage facility. Alternatives (4), reducing the plant output, and (5), shutdown of the facility, would both entail substantial additional expense for replacement electrical energy.

Table 1 presents a summarized comparison of the alternatives, in the order presented in Subsections 6.1 through 6.5. From inspection of the table, it can be seen that the most cost effective alternative is the proposed SFP modification, which is included as alternative 6. The SFP modification would provide the required storage capacity, while minimizing environmental effects, capital cost and resources committed. The staff therefore concludes that expansion of the JAFNPP SFP storage capacity is superior to the alternatives available or likely to become available within the necessary time frame.

6.5 Reduced Plant Output

Nuclear plants are usually base-loaded because of their lower costs of generating a unit of electricity compared to other thermal power plants on the system. Therefore, reducing the plant output to reduce spent fuel generation is not an economical use of the resources available. The total production costs remain essentially constant, irrespective of plant output. Consequently, the unit cost of electricity is increased proportionately at a reduced plant output. If the plant is forced to substantially reduce output because of spent fuel storage restriction, the licensee would be required to purchase replacement power or operate its higher cost fossil-fired units, if available, without any accompanying environmental advantage. The cost of electricity would therefore be increased without any likely reduction of environmental impact.

6.6 Shutdown of Facility

Storage of spent fuel from JAFNPP in the existing racks is possible but only for a short period of time. As discussed above, if expansion of the SFP capacity is not approved and if an alternate storage facility is not located, the licensee would have to shut down the unit in 1982 due to a lack of spent fuel storage facilities, resulting in the cessation of at least 821 Mwe net electrical energy production.

The incremental cost for providing replacement power if both units were shutdown would be approximately $140 million per year. This would be the cost of increased use of coal-fired and oil-fired generating facilities and the purchase of some replacement power from other utilities. This does not reflect that the licensee's investment would be idle and that JAFNPP would have to be maintained in standby or decommissioned.

## 7.0 Evaluation of Proposed Action

## 7.1 Unavoidable Adverse Environmentl Impacts

### 7.1.1 Radiological Impacts

As discussed in Section 4.0, expansion of the storage capacity of the SFP will not create any significant additional radiological effects. The additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.001 mrem/yr and 0.005 man-rem/yr, respectively. These exposures are small compared to the fluctuations in the annual dose this population receives from background radiation. The population exposure represents an increase of less than 0.1% of the exposures from the plant evaluated in the FES. The occupational radiation exposure of workers during removal of the present storage racks and installation of the new racks is estimated by the licensee to be about 6 man-rem. This is a small fraction of the total man-rem burden from occupational exposure at the plant. Operation of the plant with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the present total annual occupational exposure at this facility.

## 7.2 Relationships Between Local Short Term Use of Man's Environment and the Maintenance and Enhancement of Long Term Productivity

Expansion of the SFP storage capacity would permit more efficient use of the land already committed to this purpose. There would be no other significant changes from the evaluation in the FES.

## 7.3 Irreversible and Irretrievable Commitments of Resources

### 7.3.1 Water, Land and Air Resources

The proposed action will not result in any significant change in the commitments of water, land and air resources as identified in the FES. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings between fuel assemblies.

TABLE 1

COMPARISON OF ALTERNATIVES

| | Alternative | Cost | Benefit |
|---|---|---|---|
| 1. | Reprocessing of Spent Fuel | >$10,000/assembly | Continued production of electrical energy by JAFNPP. This alternative is not available either now or in the foreseeable future. |
| 2a. | Storage at Reprocessor's Facility | $3,000 to $6,000/ assembly per yr* plus shipping costs of $12,000 per assembly. | Continued production of electrical energy by JAFNPP. This alternative is not available now or in the foreseeable future. |
| 2b. | Storage at a new Independent Facility | $20,000-$40,000/assembly plus operating and transportation costs, and environmental impacts related to development of a new facility | Continued production of electrical energy by JAFNPP. This alternative could not be available in time to meet the present storage needs of the JAFNPP. |
| 3. | Storage at Other Nuclear Plants | Costs of shipment to other facility plus cost for subsequent shipment to an ISFSI; increased environmental costs of extra shipping and handling. | Continued production of electrical energy. However, this alternative is unlikely to be available. |
| 4. | Reduction in Plant Output | See below for replacement electricity costs. Amount of replacement required would be equivalent to at least 50% reduction in rated output for JAFNPP | Continued production of electrical energy by JAFNPP, but at much higher unit cost. The generation of replacement electricity elsewhere would probably create no less impact. |

*Since NFS and MO are not accepting fuel for storage, the cost range reflects prices that were quoted in 1972 to 1974.

TABLE 1

COMPARISON OF ALTERNATIVES

| Alternative | Cost | Benefit |
| --- | --- | --- |
| 5. Reactor Shutdown | Increased electric production expenses are estimated to be approximately $140 million/yr if JAFNPP is shut down, plus the costs of maintenance and security of the plant. | Environmental impacts associated with plant operation would cease but the generation of replacement electricity elsewhere would probably create no less impact. |
| 6. Increased Storage Capacity of JAFNPP SFP | $6,400/added assembly storage space | Continued production of electrical energy by JAFNPP. |

7.3.2 Material Resources

The proposed modification will require the utilization of about $2.6X10^5$ lb of aluminum and $4X10^4$ lb of boron carbide. The amount of aluminum and boron carbide used anually in the United States is about $10^{10}$ and $10^6$ lb respectively. The quantities required for the racks is a small amount of these resources consumed annually in the United States and is insignificant. No other significant material resources will be required because the design of the fuel pool will remain unchanged.

8.0 Benefit-Cost Balance

This section summarizes and compares the cost and the benefits resulting from the proposed modification to those that would be derived from the selection and implementation of each alternative. Table 1 presents a tabular comparison of these costs and benefits. The first three alternatives are not possible at this time or in the foreseeable future except on a short term emergency basis. Alternatives 4 and 5 have higher cost and no less environmental impacts than that of increasing storage capacity of JAFNPP.

From examination of the table, it can be seen that the most cost-effective alternative is the proposed spent fuel pool modification. As evaluated in the preceding sections, the environmental impacts associated with the proposed modification would not be significantly changed from those analyzed in the Final Environmental Statement for JAFNPP issued in March 1973.

9.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51. We have determined that the proposed license amendment will not significantly affect the quality of the human environment and that there will be no significant environmental impact attributable to the proposed action other than that which has already been predicted and described in the Final Environmental Statement for JAFNPP dated March 1973. Therefore, the staff has found that an environmental impact statement need not be prepared, and that pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Dated: June 18, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 55 to Facility Operating License No. DPR-59, issued to the Power Authority of the State of New York (the Licensee), which revised the Technical Specifications for operation of the James A. FitzPatrick Nuclear Power Plant (the facility) located in Oswego County, New York. The amendment is effective as of the date of issuance.

This amendment will allow an increase in the spent fuel storage capability up to a maximum of 2244 fuel assemblies by use of high density spent fuel racks.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Consideration of Proposed Modification to Facilities Spent Fuel Storage Pool in connection with this action was published in the Federal Register on September 12, 1978 (43FR40580). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal of the action being authorized and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility.

For further details with respect to this action, see (1) the application for amendment dated July 26, 1978, as supplemented by letters dated May 15, June 22, September 25, October 10, and November 29, 1979, April 1, April 31, and October 31, 1980, (2) Amendment No. 55 to License No. DPR-59, (3) the Commission's concurrently issued Safety Evaluation, and (4) the Commission's concurrently issued Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C., and at the Penfield Library, State University College at Oswego, Oswego, New York 13126. A single copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 18th day of June 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing