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Distribution
Docket
ORB #3
Local PDR
NRC PDR
VStello
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JShea
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Attorney, OELD
OI&E (5)
BJones (4)
BScharf (10)
JMcGough
BHarless
DEisenhut
ACRS (16)
OPA (Clare Miles)
DRoss
TBAbernathy
JRBuchanan

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 22 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment consists of changes to the Technical Specifications and is in response to your application dated March 18, 1976 and supplements dated August 11, 1976, November 30, 1976, January 18, 1977 and February 23, 1977.

The amendment consists of changes in the Technical Specifications that will increase the spent fuel pool storage capacity from 840 to 1800 fuel assemblies. The increase will: (1) provide storage for all spent fuel assemblies removed from the core between the present time and 1984, (2) provide sufficient additional fuel assembly storage capacity that the entire core (560 fuel assemblies) can be removed from the reactor vessel and stored in the spent fuel pool and (3) continue to accommodate one fuel assembly shipping cask for offsite shipping of spent fuel assemblies from the Oyster Creek spent fuel pool when offsite spent fuel shipment is resumed at some indefinite future date within the next 8 years.

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

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures and ccs:
See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE ➤	ORB #3	ORB #3	OELD	ORB #3		
SURNAME ➤	*CParrish	*Shea/Verrell	*	GLear		
DATE ➤	3/14/77	3/14/77 mJf	3/24/77	3/30/77		

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OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	CParrish CP	Shea/Verrelli mjf	MWb	GLear		
DATE	3/14/77	3/14/77	3/24/77	3/ /77		

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☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Enclosures:
1. Amendment No. 22 to License DPR-16
2. Environmental Impact Appraisal
3. Safety Evaluation
4. FEDERAL REGISTER Notice and
Negative Declaration

cc:

G. F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
Barr Building
910 17th Street, N. W.
Washington, D. C. 20006

Steven P. Russo, Esquire
248 Washington Street
P. O. Box 1060
Toms River, New Jersey 08753

Jersey Central Power & Light Company
ATTN: Mr. Thomas M. Crimmins, Jr.
Safety and Licensing Manager
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

Anthony Z. Roisman, Esquire
Roisman, Kessler and Cashdan
1025 15th Street, N. W.
5th Floor
Washington, D. C. 20005

Honorable Joseph W. Ferraro, Jr.
Deputy Attorney General
State of New Jersey
Department of Law & Public Safety
Consumer Affairs Section
1100 Raymond Boulevard
Newark, New Jersey 07102

Mark L. First
Deputy Attorney General
State of New Jersey
Department of Law & Public Safety
Environmental Protection Section
36 West State Street
Trenton, New Jersey 08625

Jersey Central Power & Light Company
ATTN: Mr. Joseph Carroll
Plant Superintendent
Oyster Creek Plant
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Chief, Energy Sys. Analysis Br. (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

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

John Russo
Bureau Chief
Bureau of Radiation Protection
380 Scotts Road
Trenton, New Jersey 08628

Ocean County Library
Brick Township Branch
401 Chambers Bridge Road
Brick Town, New Jersey 08723

Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 22
License No. DPR-16

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

   A. The application for amendment by Jersey Central Power and Light Company (the licensee) dated March 18, 1976 with supplements dated August 11, 1976, November 30, 1976, January 18, 1977 and February 23, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 22, 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

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 30, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 22

TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace page 5.3-1 with the attached revised page bearing the same number. Changed areas on the revised page are indicated by marginal lines. Also, add the attached new pages 5.3-2 and 5.3-3.

5.3 AUXILIARY EQUIPMENT

5.3.1 Fuel Storage

A. Normal storage for unirradiated fuel assemblies is in critically-safe new fuel storage racks in the reactor building storage vault; otherwise, fuel shall be stored in arrays which have a K_{eff} less than 0.95 under optimum conditions of moderation or in NRC-approved shipping containers.

B. The spent fuel shall be stored in the spent fuel storage facility which shall be designed to maintain fuel in a geometry providing a K_{∞} less than or equal to 0.95.

C. The maximum U-235 loading in grams of U-235 per axial centimeter of fuel shall not exceed 15.6 gms U-235/cm.

D. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility.

E. **The 30 ton spent fuel shipping cask shall not be lifted more than 6 inches** above the top plate of the cask drop protection system. Vertical limit switches shall be operable to assure the 6 inch vertical limit is met when the cask is above the top plate.

F. **The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F.**

BASIS

The specification of $K_{\infty} \leq 0.95$ and the maximum U-235 loading of ≤ 15.6 gm U-235/cm per axial centimeter for fuel in the spent fuel storage facility assures an ample margin from criticality. Conservative assumptions and allowance for tolerances, void effects, calculational uncertainties, pool temperature effects, etc. have been considered in the derivation of these limits (1,2). Note that the 15.6 gm U-235/cm is equivalent to a 3 w/o enrichment.(7)

The 15.6 gm U-235/cm is the limit of U-235 at any plane through the assembly perpendiculer to the length of the assembly. It is to assure that possible non-uniform enrichments along the length of fuel rods cannot lead to a critical condition.

The effects of a dropped fuel bundle onto stored fuel in the spent fuel storage facility has been analyzed. This analysis shows that the fuel bundle drop would not cause doses resulting from ruptured fuel pins that exceed 10 CFR 100 limits (3,4,5) and that dropped waste cans will not damage the pool liner.

The elevation limitation of the spent fuel shipping cask to no more than 6 inches above the top plate of the cask drop protection system prevents loss of the pool integrity resulting from postulated drop accidents. An analysis of the effects of a 100 ton cask drop from 6 inches has been done (6) which showed that the pool structure is capable of sustaining the loads imposed during such a drop. Limit switches on the crane restrict the elevation of the cask to ≤ 6 inches when it is above the top plate.

Detailed structural analysis of the spent fuel pool was performed using loads resulting from the dead weight of the structural elements, the building loads, hydrostatic loads from the pool water, the weight of fuel and racks stored in the pool, seismic loads, loads due to thermal gradients in the pool floor and walls, and dynamic load from the cask drop accident. Thermal gradients result in two loading conditions; normal operating and the accident conditions with the loss of spent fuel pool cooling. For the normal condition, the containment air temperature was assumed to vary between 65°F and 110°F while the pool water temperature varied between 85°F and 125°F. The most severe loading from the normal operating thermal gradient results with containment air temperature at 65°F and the water temperature at 125°F. Air temperature measurements made during all phases of plant operation in the shutdown heat exchanger room, which is directly beneath part of the spent fuel pool floor slab, show that 65°F is the appropriate minimum air temperature. The spent fuel pool water temperature will alarm in the control room before the water temperature reaches 120°F.

Results of the structural analysis show that the pool structure is structurally adequate for the loadings associated with the normal operation and the condition resulting from the postulated cask drop accident (9). The fuel pool floor framing was found to be capable of withstanding the maximum postulated thermal transient for at least 15 hours without exceeding ACI Code requirements. The floor framing was also found to be capable of withstanding the steady state thermal gradient conditions with the pool water temperature at 150°F without exceeding ACI Code requirements. Studies show that the critical elements of the walls indentified in the analyses of (8) are capable of withstanding eight hours of the maximum postulated thermal transient without exceeding ACI Code requirements and they are judged able to continue full functional capability for at least 10 hours under these conditions (9). The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F (9).

Since the cooled fuel pool water returns to the pool at the bottom of the pool and the heated water is removed from the surface of the pool, temperature measurement at the pool surface is appropriate to estimate the pool bulk temperature.

References

1. Amendment No. 78 to the Facility Description and Safety Analysis Report (Section 3)

2. Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report (Questions 14-20, 24, 25)

3. Amendment No. 78 to the FDSAR (Section 7)

4. Supplement No. 1 to Amendment No. 78 to the FDSAR (Question 12)

5. Supplement No. 1 to Amendment No. 78 of the FDSAR (Question 40)

6. Supplement No. 1 to Amendment No. 68 of the FDSAR.
7. Supplement No. 1 to Amendment No. 78 of the FDSAR (Question 18).
8. Addendum No. 2 to Supplement No. 1 to Amendment No. 78 of the FDSAR (Questions 5 and 10).
9. Revision No. 1 to Addendum 2 to Supplement No. 1 to Amendment No. 78 of the FDSAR (Questions 5 and 10)

ENVIRONMENTAL IMPACT APPRAISAL BY THE
DIVISION OF OPERATING REACTORS
SUPPORTING AMENDMENT NO. 22 TO DRP-16
JERSEY CENTRAL POWER AND LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION
DOCKET NO. 50-219

I. Description of Proposed Action

In their submittal of January 30, 1976, supplemented by letters dated March 18, 1976, August 11, 1976, November 30, 1976 and February 23, 1977, Jersey Central Power and Light Company (the licensee) requested approval of the NRC for an amendment to Facility Operating License No. DPP-16 and a concomitant change to the Technical Specifications for the Oyster Creek Nuclear Generating Station. This amendment to the license and change to the Technical Specifications concerns the proposed expansion of the capacity of the spent fuel storage pool (SFP).

The modification evaluated in this environmental impact appraisal is the proposal by the licensee to replace the existing fuel storage racks with closer spaced racks. The rack spacing would be changed from 11 by 6.5 inches to a nominal 9.7 x 5.9 inch center to center. The new racks would increase the storage capacity of the SFP from the present 840 fuel assemblies to 1800 fuel assemblies. Under the proposed modification, the 42 existing racks, which can hold 20 spent fuel assemblies per rack, would be replaced with 61 racks, 38 of which will hold 28 assemblies per rack and 23 of which will hold 32 assemblies per rack. The new 28 element racks will occupy the same space envelope as the present 20 element racks. The additional storage capacity would be made available by utilizing areas now vacant in the spent fuel pool.

Since the last refueling (December 1975-February 1976), Oyster Creek does not have storage capacity in their SFP to offload a full core of 560 assemblies. There are currently 326 spent fuel assemblies stored in the pool. The proposed modification would extend the spent fuel storage capability through 1983 and maintain the capability to unload all fuel from the reactor vessel. In our evaluation we considered the impacts which may result from storing an additional 960 spent fuel assemblies in the SFP for an additional seven years.

The proposed modification will not alter the external physical geometry of the spent fuel pool or require additional modifications to the SFP cooling or purification systems. The proposed modification does not affect in any manner the quantity of uranium fuel utilized in the reactor over the anticipated operating life of the facility and thus in no way affects the generation of spent uranium fuel by the facility. The rate of spent fuel generation and the total quantity of spent fuel generated during the anticipated operating lifetime of the facility and stored in the SFP remains unchanged as a result of the proposed expansion. The modification will increase the number of spent fuel assemblies stored in the SFP and the length of time that some of the fuel assemblies will be stored in the pool.

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant in New York was shut down in 1972 for alterations and expansions; on September 22, 1976, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Service (AGNS) proposed plant is under construction in South Carolina, and this facility is not licensed to operate. The General Electric Company's (GE) Midwest Fuel Recovery Plant in Illinois is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois and the storage pool at West Valley, New York (on land owned by the State of New York and leased to NFS thru 1980) are licensed to store spent fuel. The storage pool at West Valley is not full but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractual arrangements with NFS. Construction of the AGNS receiving and storage station has been completed. AGNS has applied for - but has not been granted - a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell prior to a decision on the licensing action relating to the separation facility.

The NRC Staff is preparing a generic environmental impact statement on spent fuel storage of light water power reactor fuel and is expected to complete this statement by the fall of 1977. The proposed expansion of the SFP capacity at Oyster Creek will afford the licensee operational flexibility by providing storage space for spent fuel discharges through 1983 with storage space for an emergency full core discharge.

II. Environmental Impacts of Proposed Action

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling the storage of spent fuel from light water reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, the following five specific factors should be applied, balanced, and weighted in the context of the required environmental statement or appraisal.

a. Is it likely that the licensing action here proposed would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity?

The Oyster Creek reactor core contains 560 fuel assemblies. The facility was licensed in April 1969 and commenced commercial operation in December 1969. The Oyster Creek SFP was designed on the basis that a fuel cycle would be in existence that would only require storage of spent fuel for a year or two prior to shipment to a reprocessing facility. Therefore, a pool storage capacity for 840 assemblies (1 1/2 cores) was considered adequate. This provided for complete unloading of the reactor even if the spent fuel from two refuelings were in the pool. Typically, the Oyster Creek Nuclear Generating Station is refueled once a year. Each refueling replaces about one-quarter of the core (about 140 assemblies) and each new assembly contains about 175 kilograms of uranium.

Jersey Central Power and Light Company had a contractual agreement with Nuclear Fuel Services (NFS) whereunder the licensee has shipped 224 spent fuel assemblies to NFS's reprocessing plant in West Valley, New York for storage. The contractual arrangements were fulfilled in 1975, the last year in which Oyster Creek shipped out spent fuel. No other shipping arrangements have been made by the licensee. On September 22, 1976, NFS announced that they were withdrawing from the fuel reprocessing business. There are currently 326 spent fuel assemblies stored in the Oyster Creek SFP. With the existing storage racks, full core discharge is no longer possible. If about 140 fuel assemblies are discharged each year, the SFP will be filled after the Spring 1979 refueling.

Since spent fuel reprocessing facilities cannot assuredly be available to Jersey Central Power and Light Company prior to the mid-1980's (and, therefore, no spent fuel can be shipped for reprocessing), spent fuel discharges subsequent to 1979 will have to be stored or the facility shut down. The proposed licensing action (i.e., installing new racks of a design that permits storing more assemblies in the same space) would provide the licensee with additional operating flexibility which is desirable even if adequate offsite storage facilities hereafter become available to the licensee.

We have concluded that a need for additional spent fuel storage capacity exists at the Oyster Creek Nuclear Generating Station which is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity.

b. Is it likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?

With respect to this proposed licensing action, we have considered commitment of both material and nonmaterial resources. The material resources considered are those to be utilized in the expansion of the SFP. The proposed fuel rack modification will involve removing the old racks and replacing them with racks which have a closer center-to-center spacing of the fuel assemblies.

Under the proposed modification, the present spent fuel racks will be replaced by new spent fuel racks that will increase the storage capacity to 1800 assemblies. The new spent fuel rack is a modular design with fuel arranged in slabs. All material used in the racks is Type 304 stainless steel. There will be two types of rectangular boxes fabricated of 0.090 inch thick sheet. One of the boxes will be sized to hold two fuel assemblies in a close packed condition while the other will hold water. The box array is joined by welding to form a solid honeycomb structure. When these racks are installed in the fuel pool, there will be rows of close packed fuel assemblies separated by 3.6 inch wide water boxes.

The total quantity of stainless steel to be utilized in the new spent fuel racks is approximately 300,000 pounds. The racks do not use a poison material such as boron impregnated stainless steel, B_4C plates or boral. The amount of stainless steel used annually in the U. S. is about 2.82×10^{11} lbs. The material is readily available in abundant supply. The amount of stainless steel required for fabrication of the new racks is a small amount of this resource consumed annually in the United States. We conclude that the amount of material required for the racks at Oyster Creek is insignificant and does not represent an irreversible commitment of natural resources. This licensing action would not constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or other actions that might be taken by the industry in the future to alleviate fuel storage problems. No other resources need be allocated because the other design characteristics of the SFP remain unchanged. 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 among fuel assemblies.

The increased storage capacity at the Oyster Creek spent fuel pool was considered as a nonmaterial resource and was evaluated relative to proposed similar licensing actions within a one year period (the time we estimate is necessary to complete the generic environmental statement) at other nuclear power plants, fuel reprocessing facilities and fuel storage facilities. We have determined that the proposed expansion in the storage capacity of the SFP is only a measure to allow for continued operation and to provide operational flexibility at the facility, and will not affect similar licensing actions at other nuclear power plants.

We conclude that the expansion of the spent fuel pool at the Oyster Creek Nuclear Generating Station prior to the preparation of the generic statement does not constitute a commitment of either material or non-material resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing action designed to ameliorate a possible shortage of spent fuel storage capacity.

c. Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overlooking any cumulative environmental impacts?

The SFP at Oyster Creek was designed principally to store spent fuel assemblies prior to shipment to a reprocessing facility. These assemblies may be transferred from the reactor core to the SFP during a core refueling, or to allow for inspection and/or modification to core internals. The latter may require the removal and storage of up to a full core. The assemblies are initially intensely radioactive due to their fission product content and have a high thermal output. Thus they are stored in the SFP to allow for radioactive and thermal decay. The major proportion of decay occurs during the 150 day period following removal from the reactor core. After this period, the assemblies may be withdrawn and placed into a heavily shielded fuel cask for offsite shipment. Space permitting, the assemblies may be stored for more than 150 days in the SFP, allowing continued fission product decay and thermal cooling prior to shipment from the facility.

Potential impacts, nonradiological and radiological, relative to the construction and operation of the expanded SFP at this facility were considered by the NRC Staff. No environmental impacts on the environs outside the spent fuel storage building were identified that would be associated with the proposed construction of the expanded SFP. The impacts within this building are expected to be limited to those normally associated with metal working activities.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be an additional discharge of heat to Barnegat Bay. Storing spent fuel in the SFP for a longer period of time will add more heat to the SFP water. Part of this heat is transferred to the Bay through several intermediary cooling water system.

The Final Environmental Statement (FES) related to the operation of the Oyster Creek Nuclear Generating Station was issued December 1974. As discussed below, the storage of spent fuel on-site for a longer period of time will not significantly change the environmental impacts evaluated in the FES.

Both the licensee and the staff have evaluated the existing SFP cooling system and have concluded that the latter has adequate capacity to maintain the pool water temperature below 125°F with the normal refueling schedule (i.e., annual replacement of 1/4 of the core). The two SFP heat exchangers are cooled by the Reactor Building Closed Cooling Water System which is in turn cooled by the service water system. Compared to the existing heat load on the Reactor Building and the Turbine Building Closed Cooling Water Systems and the total heat rejected to Barnegat Bay by the once-through circulating water system, the small additional heat load from the SFP cooling system (attributable to the longer storage of additional spent fuel) will be negligible.

The only potential offsite radiological environmental impact associated with this expansion would be an increment in the long-lived radioactive effluents (Kr-85) released from the facility and this has been determined to be environmentally insignificant. The expansion of the SFP will allow spent fuel to be stored for an additional sevenyear period without shipment offsite and still maintain space to off-load a full core.

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 released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are predominantly nonvolatile and, as with the activated corrosion product nuclides, the primary impact 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.

To provide redundancy and the ability to off-load a full core earlier (i.e., 10 days) than if the Spent Fuel Pool Cooling System (SFPCS) were not modified, Jersey Central Power and Light will install two new full capacity pumps and heat exchangers in parallel with the existing pumps and heat exchangers. The existing SFPCS consists of a single loop containing two pumps, two heat exchangers, a 150 cu. ft. mixed bed demineralizer and a back-flushable mixed resin precoat filter. The pumps and heat exchangers are located in the reactor building. The fuel pool filter and demineralizer, which become radioactive as they collect corrosion and fission product nuclides, are located in the radwaste building.

The fuel pool cooling system circulates, filters, and demineralizes the water in the fuel pool during plant operation, and in the reactor cavity, the equipment storage cavity, and the fuel pool during refueling. This is done to maintain clear water and to minimize the amount of crud and corrosion products in the water. Normal flow rate through the demineralizer and/or filter is 400 gpm. Operating experience shows that the fuel pool water quality can generally be maintained by the fuel pool filter alone.

Conductivity is maintained at less than 1.0 μmho/cm and undissolved solids less than 0.5 ppm. The fuel storage pool water temperature and quality are thus equivalent to reactor water conditions. The reactor cavity water and the fuel pool water circulate together when the fuel pool gates are open during refueling. At that time, the shutdown cooling system is also operated continuously.

Fuel pool water flows over weirs through two surface skimmers, both at the north side of the pool into surge tanks which have a normal level below the pool level. The pool water is pumped from the surge tanks through heat exchangers, a filter, a demineralizer, and returned to the fuel pool through two return diffusers at the bottom of the pool in the southwest and southeast corners.

During refueling, the reactor cavity is filled and the gates removed between the pool and the reactor cavity. Water flows over weirs, through four surface skimmers distributed around the reactor cavity and through six surface skimmers distributed around the equipment storage cavity, then joins the flow from the pool into the surge tanks. Return flow goes into the reactor cavity through two return diffusers mounted on the cavity wall above the reactor flange.

Storing additional spent fuel in the SFP may increase the amount of corrosion and fission product nuclides introduced into the SFP water. The purification system is capable of removing the increased radioactivity so as to maintain acceptable radiation levels above and in the vicinity of the pool. Redesign of the SFP racks increases only the storage capacity of the pool and not the frequency or the amount of the core to be replaced for each fuel cycle. Thus, the amount of corrosion product nuclides released into the pool during any year will be about the same regardless of the length of time or number of assemblies stored in the pool. Expansion of the capacity could increase the potential for increasing the amount of fission products introduced into the SFP water. Experience indicates that there is little radionuclide leakage from spent fuel stored in pools. The leakage of radionuclides from the fuel is greatly reduced after the fuel has cooled for several weeks. The predominance of radionuclides in the spent fuel pool water appears to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with the water in the spent fuel pool during refueling operations) or crud dislodged from the spent fuel during transfer. 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 reactor operating conditions of approximately 800°F. A few days after refueling, the spent fuel cools in the spent fuel pool so that the fuel rod temperature is relatively cool, approximately 180°F. This substantial temperature reduction reduces the rate of release of fission products from the fuel pellets and decreases the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the cladding. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. According to the owners, there has never been indication of leakage of fission products from spent fuel stored in the Midwest Fuel Recovery Plant (MFRP) at Morris, Illinois, or at Nuclear Fuel Services' (NFS) storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite spent fuel pool, this fuel was later shipped to either MFRP or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no detectable leakage from this fuel in the offsite storage facility.

The licensee does not expect to change the frequency of operation of the SFP purification system as a result of the fuel storage rack modification. The demineralizer is currently changed on the basis of conductivity in the effluent. The filter is presently backwashed on a monthly basis or in the event of high pressure drop and this is not expected to change. On the above basis, the licensee estimates that the modified SFP is not expected to generate a significantly higher quantity of solid radwaste. To upperbound any potential increase in solid waste, we have assumed that the amount of solid radwaste may be increased by an additional resin bed a year. During 1975, a total of 34,319 cubic feet of solidified waste was shipped offsite in 162 shipments. If the increased storage of spent fuel does increase the amount of solid waste by 150 cubic feet per year, the increase in total waste volume would be less than 1% and would not have any significant additional environmental impact.

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 and by utilizing realistic assumptions for radionuclide concentrations in the SFP water and for occupancy times. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the 23 foot depth of water shielding the fuel.

The Oyster Creek SFP is being utilized temporarily as a storage area for some high level radioactive waste such as Local Power Range Monitors (LPRM's) and channel clips. These sources increase the dose rates above the surface of the pool and thus the occupational exposure to personnel working in the spent fuel pool area. The licensee has stated that it is their intent to remove and ship the waste material now stored in the pool at the refueling outage scheduled for the Spring of 1977. After removal of the waste material, the licensee's plan for removal of the existing racks and installation of the new racks may include the use of contractor divers in addition to other contractor and plant personnel. The new racks will be added over a period of several years on an as-needed basis. The new racks can be installed while the plant is operating. Replacing the racks over a period of several years will not change the total occupational exposure or other minor environmental effects associated with the installation, but will spread the exposure over several years. The licensee has estimated the occupational exposure for replacement of the existing racks to be about 15 to 20 man-rem. We consider this a reasonable estimate. This occupational dose, and doses received from subsequent normal operations in the spent fuel pool area will represent less than two percent of the present total annual occupational exposure at this facility. Consequently, the small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

The only significant noble gas isotope remaining in the SFP and attributable to storing additional assemblies for a longer period of time would be Krypton-85. Based on operating experience for Zircaloy clad fuel (see NUREG-0017), we have assumed that 0.12% of all fuel rods will have cladding defects which permit the escape of fission product gases. This value is the weighted average percent defective fuel for nine pressurized water reactors. It is assumed that the fission product gases escape on a relatively linear basis with time. On this basis, we have conservatively estimated that an additional 16 curies per year of Krypton-85 will be released when the modified pool is completely filled. The fuel storage pool area is continuously ventilated. Normally, this air is released through the plant stack. If the plant does eventually release an additional 16 curies per year of Kr-85 as a result of the proposed modification, the increase would result in an additional offsite dose of less than 0.01 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The calculated dose to the estimated population within a 50 mile radius of the plant is less than 0.01 man-rems/year, which is also insignificant and less than the natural fluctuations in the dose this population would receive from background radiation. Thus, we conclude that the proposed modification will not have any significant impact on radiation levels or personnel exposure offsite.

Assuming that the spent fuel will be stored onsite for several years (rather than shipped offsite after 6 to 24 months storage as originally planned), Iodine-131 releases will not be significantly increased by the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between each annual refueling. Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above the 125°F used in the design analysis during normal refuelings or during a full core off-load. The licensee has proposed procedural controls which will be used to insure that a full core will not be unloaded to the spent fuel pool until it has been determined that the SFP water temperature will not exceed 125°F. The fuel pool cooling system and shutdown cooling system were originally designed with capped connections for a cross connect from the fuel pool system to the "A" heat exchanger of the shut down cooling system. This cross connect could augment the fuel pool cooling system, approximately doubling the present cooling capacity. To insure that the pool water temperature will be maintained below 125°F even when a full core is offloaded, Jersey Central Power and Light will proceed with the installation of two new full capacity pumps and one heat exchanger in parallel with the two existing pumps and heat exchangers. Since the temperature of the pool water will be maintained below 125°F, it is not expected that there will be any significant change in evaporation rates and the release of tritium as a result of the proposed modification.

We consider the licensee's cask drop protection system adequate for the prevention of cask tip accidents. The dashpot structure and fuel pool structure are adequate for loadings imposed during postulated cask tip accidents. The cask travel will be limited to the specified path and other heavy loads will not be carried over spent fuel. Further, movement of the fuel cask will not be permitted until the details of the means used to limit the height to which the cask can be raised over the operating deck have been submitted by the licensee and approved by the NRC staff. The proposed modification will not change the rate or number of spent fuel assemblies transferred from the reactor into the SFP. The consequences of spent fuel accidents therefore remain unchanged from that discussed in the FES and the probability of fuel handling accidents is not significantly increased as a result of the additional fuel transfers required during the modification of the pool.

The staff has considered the potential cumulative environmental impacts associated with the expansion of the SFP and have concluded that they will not result in radioactive effluent releases that significantly affect the quality of the human environment during either normal operation of the expanded SFP or under postulated fuel handling accident conditions.

d. Have all technical issues which have arisen during the review of this application been resolved within that context?

This impact appraisal and the accompanying safety evaluation report point out that all questions concerning health, safety and environmental concerns have been answered.

e. Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?

In regard to this licensing action, the staff has considered the following alternatives: (1) shipment of spent fuel to a fuel reprocessing facility, (2) shipment of spent fuel to a separate fuel storage facility, (3) shipment of spent fuel to another reactor site, and (4) ceasing operation of the facility. These alternatives are considered in turn.

The proposed rack modification and replacement will cost the Jersey Central Power & Light Company about 1.5 million dollars for the rack design, fabrication, and installation. While this is costly, the alternatives are more costly.

(1) Jersey Central Power and Light Company had a contractual agreement with Nuclear Fuel Services (NFS) whereunder the licensee has shipped 224 spent fuel assemblies to NFS's reprocessing plant in West Valley, New York for storage. The contractual arrangements were fulfilled in 1975, the last year in which Oyster Creek shipped out spent fuel. No other shipping arrangements have been made by the licensee. As discussed earlier, none of the three commercial reprocessing facilities in the U.S. are currently operating. The General Electric Company's Midwest Fuel Recovery Plant (MFRP) at Morris, Illinois is in a decommission 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." In their letter to NRC and letters to utilities with whom NFS had contracts for storage and reprocessing of spent fuel, NFS discussed the reasons for their decision. For several years, NFS had been seeking the licensing approval of the Commission for modifications of the reprocessing plant at West Valley to increase its operating capacity and for operation of the Modified facility. When the Commission determined that such approval would require both a construction permit and an operating license amendment, NFS filed an application for amendments to Provisional Operating License No. CSF-1, which was docketed on December 17, 1973. During the course of review of this application, new regulatory requirements were periodically identified; for example, in April 1976, the NRC staff concluded that seismic requirements would have to be significantly increased. NFS estimated that the new requirements would increase the cost of the project from the $15 million originally estimated to over $600 million and delay resumption of reprocessing until 1988. On the above basis, NFS concluded "that the project is commercially impractical in light of regulatory requirements that have arisen since the project was initiated." The Allied General Nuclear Services (AGNS) reprocessing plant received a construction permit on December 18, 1970. In October 1973, AGNS applied for an operating license for the separation facility; construction of the latter is essentially complete. On July 3, 1974, AGNS applied for a materials license to receive and store up to 400 MTU in spent fuel in the onsite storage pool, on which construction has been completed. Hearings are expected to be completed on the materials license application by mid 1977. However, the AGNS separations plant will not be licensed until the issues presently being considered in the GESMO proceedings are resolved and these proceedings are completed. 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 7000 MTU in spent fuel. The application for a construction permit is under review. Therefore, shipment of spent fuel to a reprocessing plant is not an available alternative for several more years.

(2) In 1975, the licensee evaluated storage at commercial storage facilities such as Nuclear Fuel Services. At that time, it was determined that the average cost, including transportation, for such storage would be approximately $3620/year/assembly, compared to the approximately $1500 per assembly cost of modifying the present SFP. At present, it is uncertain whether firm contractual arrangements could be made with any existing reprocessing facility to store additional spent fuel. An alternative to expansion of onsite spent fuel pool storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. An ISFSI could be designed using dry storage technology. Fuel storage pools as GE Morris and NFS are functioning as ISFSIs although this was not the original design intent. Likewise, if the receiving and storage station at AGNS is licensed to accept spent fuel, it would be functioning as an ISFSI until the separations facility is licensed to operate. The license for the GE facility at Morris, Illinois was amended on December 3, 1975 to increase the storage capacity to about 750 MTU; approximately 200 MTU is now stored in the pool. The NFS facility has capacity for about 260 MTU, with approximately 170 MTU presently stored in the pool. However, since NFS withdrew from the fuel reprocessing business, they are not at present accepting additional spent fuel for storage even from those reactor facilities with which they had contracts. The AGNS will have capacity for about 400 MTU if they are licensed to receive spent fuel.

With respect to construction of new ISFSIs, Regulatory Guide 3.24, "Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation," issued in December 1974, recognizes the possible need for ISFSIs and provides recommended criteria and requirements for water-cooled ISFSIs. Pertinent sections of 10 CFR Part 19, 20, 30, 40, 51, 70, 71 and 73 would also apply.

It is estimated that at least five years would be required for completion of an independent fuel storage facility. This estimate assumes one year for preliminary design; one year for preparation of the license application, Environmental Report, and licensing review in parallel with one year for detail design; two and one-half years for construction and receipt of an operating license; and one-half year for plant and equipment testing and startup.

Industry proposals for 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 joing 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. In 1974, E. R. Johnson Associates estimated their construction cost at approximately $9000 per spent fuel assembly. At this rate, it would cost the licensee over $8,000,000 to store the additional 960 spent fuel assemblies that the proposed modification will accommodate, plus there would be additional costs for shipment and safeguarding the fuel. On December 2, 1976, Stone and Webster Corporation submitted a topical report requesting approval for a standard design for an independent spent fuel storage facility. No specific locations were proposed, although the design is based on location near a nuclear power facility. No estimated costs for fuel storage were included in the topical report. An independent spent fuel storage installation is not a viable alternative based on cost or availability in time to meet the licensee's needs. It is also unlikely that the total environmental impacts of constructing an independent facility and shipment of spent fuel would be less than the minor impacts associated with the proposed action.

(3) Consideration was given to possible storage in the spent fuel pool of the Metropolitan Edison Company's Three Mile Island Unit (TMI-1), a PWR facility. The Metropolitan Edison Company is a sister subsidiary of Jersey Central in the General Public Utilities Corporation. To do this, it is estimated that the needed modification to the PWR storage racks of TMI-1 would cost $1.2 million and $2,000/ assembly for shipping. Only about 150 assemblies could be shipped before this alternative loses its economic advantage. Additionally, impact upon future storage capacity for TMI-1 also weighs against this decision.

The alternative of storing spent fuel in the storage pool of another nuclear reactor also compares poorly with the proposed action. The cost probably would be comparable to the cost of storage at a commercial storage facility and the licensee would be utilizing storage space which the recipient might require at a future date. Such a transfer would also impose additional fuel handling and transportation requirements and related additional shipping expense.

According to a survey conducted and documented by the Energy Research and Development Administration, as many as 46 percent of the operating nuclear power plants will lose the ability to refuel during the period 1975-1984 should there not be any additional spent fuel storage pool expansions or commitments to utilize offsite storage facilities. Thus, the licensee cannot assuredly rely on any other power facility to provide additional storage capability except on a short-term emergency basis.

Because the fuel reprocessing problem is generic to the nuclear industry, it is not logical to store fuel from the Oyster Creek Nuclear Generating Station at another facility. In the long-term, other facilities will have no more storage space available than Oyster Creek has itself.

(4) Typically, the Oyster Creek Nuclear Generating Station is refueled once a year. Each refueling normally replaces about one quarter of the core (140 assemblies). The present storage capacity of the SFP is 840 fuel assemblies; however, there are presently 326 assemblies stored in the pool from previous refuelings. Thus, Oyster Creek cannot offload a full core of 560 assemblies, although removal of the entire core will be necessary if the licensee is to proceed with inspection of certain reactor internals as now tentatively planned during the Spring 1977 refueling outage. Even if offload of a full core was not required, with annual discharges the existing storage capacity of the spent fuel pool would be filled by the discharge expected in the Spring of 1979. This implies that Oyster Creek would be unable to discharge spent fuel in 1980 and that operation of the Station would have to be terminated. The current energy replacement value for Oyster Creek is approximately $360,000 a day (assuming 620 MWe), and is not an economic alternative.

In summary, alternatives (1) to (3) described above do not offer the operating flexibility of the proposed action nor could they be completed as rapidly as the proposed action. The alternatives of shipping the spent fuel to a reprocessing facility, an independent storage facility or to another reactor would be more expensive than the proposed action and might preempt storage space needed by another utility. The alternative of ceasing operation of the facility would be more expensive than the proposed action because of the need to provide fossil fuel replacement power. In addition to the economic advantages of the proposed action, we have determined that the expansion of the SFP would have a negligible environmental impact. Accordingly, deferral or severe restriction of the proposed action would result in substantial harm to the public interest.

III. 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 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6 and have applied, weighted, and balanced the five factors specified by the Nuclear Regulatory Commission in 40 CFR 42801. We have determined that the license amendment will not significantly affect the quality of the human environment. Therefore, the Commission 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 22 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

JERSEY CENTRAL POWER AND LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

Introduction

By letter dated March 18, 1976, the Jersey Central Power & Light Company (JCP&L) submitted an application for an amendment to Appendix A of Provisional Operating License No. DPR-16 to increase the spent fuel pool storage capacity of the Oyster Creek Nuclear Generating Station from 840 to 1800 fuel assemblies. Supplemental information in response to NRC letter dated June 24, 1976 was provided by JCP&L in letters dated August 11, 1976, November 30, 1976, January 18, 1977, and February 23, 1977. Notice of Proposed Issuance of an amendment to Provisional Operating License No. DPR-16 issued to JCP&L was published in the FEDERAL REGISTER on April 22, 1976 (41 FR 16891).

Discussion

The spent fuel pool at the Oyster Creek Nuclear Generating Station contains 326 spent fuel assemblies at the present time. Spent fuel has been stored in the pool since the first core refueling following plant startup on December 23, 1969. Prior to January 1976, 224 of the oldest spent fuel assemblies that had been stored in the pool were shipped from the site. There are no plans at this time for additional offsite shipments during the next few years. Since there is storage space for only 840 fuel assemblies and since the core contains 560 fuel assemblies, the Oyster Creek facility cannot, with the existing spent fuel storage racks, accommodate removal and storage in the spent fuel pool of all of the fuel assemblies in the core.

The proposed increase in spent fuel storage capacity from 840 fuel assemblies will (1) provide storage for all spent fuel assemblies removed from the core between the present time and 1984, (2) provide sufficient additional fuel assembly storage capacity that the entire core (560 fuel assemblies) can be removed from the reactor vessel and stored in the spent fuel pool and (3) continue to accommodate one fuel assembly shipping cask for offsite shipping of spent fuel assemblies from the Oyster Creek spent fuel pool when offsite spent fuel shipment is resumed at some indefinite future date within the next 8 years.

Our evaluation considers:

1. Structural Adequacy of the Proposed Spent Fuel Racks and Pool
2. The Potential for Unintentional Criticality
3. Spent Fuel Pool Cooling Capacity
4. Fuel Handling and Installation of the Modified Spent Fuel Racks

Evaluation

1. Structural Adequacy of the Proposed Spent Fuel Racks and Pool

   The proposed spent fuel pool modification consists of replacing the existing fuel storage racks with new spent fuel racks that will increase storage capacity from 840 to 1800 fuel assemblies. Each new rack assembly is made up of rectangular steel boxes with a base plate at the bottom of each box to support the fuel assemblies and holes in the base plate to permit coolant flow. A flux trap region between the fuel boxes is formed by additional rectangular water boxes. Each box in the assembly is welded to adjacent boxes to form a honeycomb box structure arrangement. Each rack assembly is mechanically joined to adjacent rack assemblies in minimum groups of twenty-four. The rack assemblies are bolted to support beams which are fastened to the bottom of the pool floor by existing swing bolts. There are no structures to connect the racks to the fuel pool walls. All material used in the fabrication and construction of the racks is type 304 stainless steel.

   All applicable structural steel items were designed to the AISC Specification for Design, Fabrication and Erection of Structural Steel for Buildings, revision 7, in conjunction with the material allowables from the 1974 ASME Boiler and Pressure Vessel Code (B&PV). The welds used to fasten the fuel and water boxes together were designed to meet Section VIII of the 1974 B&PV Code.

   The seismic design of the racks is based on the response spectra and damping values presented in the Oyster Creek FSAR. No benefit is taken for the damping effect of the water. The analyses included the mass of an external water envelope of appropriate thickness as well as the additional mass due to water trapped inside the fuel and water boxes. In the design of the racks a horizontal acceleration of 0.312g was applied simultaneously with normal gravity plus or minus a vertical acceleration of 0.312g. The direction of the horizontal seismic component was assumed to be in the worst-case direction which results in the maximum loads at any fuel rack corner joint. As an independent check on the adequacy of the design, additional calculations were performed by the licensee to demonstrate equivalence to solutions that consider seismic excitations along three orthogonal directions imposed simultaneously as recommended in Regulatory Guide 1.92.

The fuel racks and supporting structures were designed* for the extreme environmental conditions occurring simultaneously with the abnormal plant conditions (i.e., fully-loaded spent-fuel racks in a hot pool (200°F) undergoing a safe shutdown earthquake-seismic Category I). The racks were also analyzed for normal operating conditions, severe environmental conditions and extreme environmental conditions. Normal code stress limits were used as acceptance criteria for all of the above postulated load conditions. In addition, the licensee considered the loads from a dropped fuel assembly and found that the racks have adequate structural strength to withstand the effects of such an accident. We agree with these results.

The new racks, in minimum groups of 24, can be installed on an "as needed" basis because each assembly will meet seismic Category I requirements. The base supports are installed first and fastened to the pool floor by the existing swing bolts. Each rack assembly is then positioned, bolted to the base support, and finally tied to adjacent assemblies to form a minimum grouping of twenty-four racks. All existing racks in the area where a new rack is to be installed will be unloaded and the fuel placed in a remote area of the pool. Although a number of precautions will be taken to preclude the possibility of dropping a rack assembly during its installation, the fuel pool floor integrity would not be jeopardized if a rack assembly were dropped from the pool sill.

The criteria used in the analysis, design, and construction of the new spent fuel racks to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory 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.

* Quality assurance requirements for installation, inspection, and testing will be in accordance with the JCP&L Operational Quality Assurance Plan (March 19, 1976). In accordance with JCP&L policy the plan meets the requirements of the Code of Federal Regulations, 10 CFR Appendix B.

The licensee has also performed detailed analyses of the spent fuel pool structure to verify its ability to withstand the increase in overall loading as a result of the proposed fuel pool modification. The loads considered in their analysis include: the weight of the pool; other building loads; hydrostatic loads; the weight of the spent fuel elements, supporting racks, and the spent fuel cask; seismic loads; dynamic loads from a postulated cask drop accident; and thermal gradients based on pool water temperature of 125°F.

The load combinations, design and analysis procedures, and the structural acceptance criteria used in the evaluation are consistent with Section 3.8.4 of the Standard Review Plan. However, due to certain reinforcement details in the pool floor slab panels, the licensee used additional criteria to demonstrate that the pool slab can adequately transfer shear force to the supports across postulated cracks which may result from the effects of thermal loads. The additional criteria is based on the provisions of Section 11.15 of the ACI 318-71 Code and the results of experimental investigations.

In order to preclude the need for additional structural calculations at abnormally high temperatures the licensee will provide a new cooling system, in addition to the existing system, to assure that pool temperature remains below the temperature at which the alarm is set (i.e., no higher than 120°F). A change to the Technical Specifications will limit pool temperature to 125°F. In order to demonstrate the safety margin above this temperature limit, the licensee has performed analyses which conclude that the pool structure could withstand steady state pool water temperatures of at least 140°F.

We conclude that there is reasonable assurance that the spent fuel pool structure will withstand the specified design conditions without impairment of its structural integrity or the performance of required safety functions.

2. Criticality Considerations

The proposed spent fuel racks, which are designed to support the stored fuel assemblies on a nominal 9.7 x 5.9 inch pitch under safe shutdown earthquake accelerations, are to be fabricated from .090 inch thick, type 304 stainless steel. This steel will be made into two types of rectangular boxes. One of the boxes will be sized to hold two fuel assemblies in a close-packed condition, while the other will hold water to moderate and absorb neutrons. When these racks are installed in the fuel pool there will by rows of close-packed fuel assemblies separated by the 3.6 inch wide water boxes.

The licensee provided a criticality analyses for these fully loaded racks using their version of the LEOPARD computer program to get four group cross sections for the PDQ-7 diffusion theory calculations. The fuel region in the basic PDQ cell is 5.166 inches square resulting in a fuel region volume fraction of .47 for the nominal storage lattice. JCP&L reports that the criticality analyses for this array were based on an enrichment of 3.9 weight percent U^{235} and that this enrichment corresponds to a maximum fuel loading of 15.6 grams of U-235 per axial centimeter of fuel assembly.

The maximum effect of mechanical fabrication tolerances, fuel assembly positioning uncertainty, stainless steel thickness, and water temperature on the neutron multiplication factor was calculated in addition to the nomimal neutron multiplication factor for no neutron leakage (i.e., for infinite radial and axial dimensions).

For unirradiated fuel assemblies with a fuel loading of 15.6 grams of U-235 per axial centimeter of fuel assembly and no burnup poison, JCP&L calculates the infinite neutron multiplication factor, K_∞ to be 0.89. Nominal dimensions for the lattice with a 3.6 inch wide water box and a water temperature of 80°F were assumed. The nominal neutron multiplication factor for the worst case condition including a uniform increase in the water temperature to 200°F is increased by 0.02. Thus the maximum k_∞ for this storage lattice is calculated to be 0.91. The conservatism in this calculation is evident when normal spent fuel pool conditions are considered. Normally spent fuel assemblies (less 235 U) are stored in the pool after about 4 years of producing power in the core. The spent fuel 235 U enrichment is about one third of new fuel assembly 235 U enrichment. Since the criticality calculation is based on new fuel assemblies rather than spent fuel assemblies it is conservative and K_∞ is therefore even lower than 0.91.

The major uncertainties in the licensee calculations are in the accuracy of the four group cross sections and in the methods for accounting for the non-isotropic scattering of neutrons when they collide with hydrogen atoms. The accuracy of the four group cross sections was determined by using the LEOPARD & PDQ-7 programs to calculate K_{eff} for more than thirty critical experiments. Nineteen of these experiments had stainless steel in them; therefore, all of the materials in the storage lattice were included. The maximum difference between the calculated and experimentally measured neutron multiplication factors was .019 delta k/k. Allowance for this amount of reactivity uncertainty increases the calculated k_∞ from 0.91 to 0.93. In its response to our request for information on the uncertainty in the calculation for non-isotropic hydrogen scattering, JCP&L stated that a 15 percent variation in the fast group neutron diffusion coefficient caused a change of only .004 in the neutron multiplication factor and that this 15 percent change is considerably greater than the anticipated difference between diffusion and transport theory. A comparison of results of other calculations has shown that higher order transport calculations should tend to decrease the calculated neutron multiplication factor in this storage lattice. Thus, with allowance for maximum uncertainties it can be concluded that k_∞ will be equal to or less than 0.93. Since no allowance has been made for axial leakage of neutrons from the actual fuel pool geometry the K_{eff} of the stored fuel will be less than 0.93 and will meet the criterion of our review plans of $K_{eff} \leq 0.95$ with a margin equal to or greater than 0.02 in multiplication factor.

A potentially significant increase in neutron multiplication factor in this array of stored fuel assemblies could be obtained by somehow displacing the water in the water boxes with trapped air or steam while the fuel assemblies are filled with water. In response to this expressed concern the licensee states and we agree that:

> "For all lead-in guides, the major flow restriction is the bottom plate holes. There is no way that sufficient crud can build up to obstruct either the 3/4" hole (bottom) or lead-in guide openings due to the large flow area provided."

Also, since the 3/4" diameter holes in the bottom plates should act as a filter to catch any conceivable object before it has a chance to plug up the top of the water boxes, we find that when the fuel boxes are filled with water, steam or air will not be trapped in the water boxes. Therefore, the margin to criticality remains below the NRC acceptable value of $K_{eff} \leq 0.95$.

We conclude that when any number of fuel assemblies, which have no more than 15.6 grams of U-235 per axial centimeter of assembly, are loaded into the spent fuel pool racks modified as proposed that the neutron multiplication factor will be ≤ 0.93. Since this is less than the NRC's acceptance criterion of $K_{eff} = 0.95$ we find the proposed design to be acceptable.

On this basis, we conclude that the Technical Specification changes to prohibit the storage of fuel assemblies that contain more than 15.6 grams of U-235 per longitudinal centimeter of assembly are acceptable and there is reasonable assurance that the health and safety of the public will not be endangered by the use of these racks.

3. <u>Spent Fuel Cooling</u>

JCP&L has reported that the spent fuel pool cooling system for the Oyster Creek Nuclear Generating Station is designed to remove one thermal megawatt of decay heat from spent fuel assemblies stored in the pool for every 17°F difference between the temperature of the fuel pool outlet water and the temperature of the cooling water (in this case the cooling water is the water in the Reactor Building Closed Cooling Water System). The heat sink temperature of the Oyster Creek plant is in the range of 40°F to 90°F depending on the time of the year. JCP&L also noted that the design temperature for the fuel pool outlet water temperature is 125°F. Calculations show that a pool temperature of 140°F can be tolerated, but 125°F has been established as the limit for normal operation to identify a conservative temperature safety margin. An alarm will annunciate in the control room if the fuel pool surface temperature exceeds 120°F.

In regard to the maximum heat load on the spent fuel pool cooling system, JCP&L calculated the decay heat for a full core discharge to the fuel pool ten days after shutting down the reactor with nine 1/4 core reload batches already in the pool. (Ten days is the minimum time necessary to unload the core into the spent fuel pool and replace the gate between the spent fuel pool and reactor cavity.) The calculated maximum heat load is 5.5 thermal megawatts (MWt) with 95% of the heat load from the full core and the last two 1/4 cores to be unloaded. For the normal refueling offload of 1/4 core (140 fuel assemblies) with twelve 1/4 cores already in the pool, JCP&L calculated the spent fuel pool heat load ten days after reactor shutdown to be 1.73 MWth.

The calculated water temperature of the pool as a function of time following a complete loss of spent fuel pool cooling capability shows that it would take at least 9.5 hours for boiling to occur if the initial pool temperature was 140°F with a core off loaded into the pool and all of the spent fuel pool racks filled.

We have calculated, using the total decay energy curve of the NRC Standard Review Plan, "Technical Position APCSB 9-2", a value of 5.02 MWt of decay heat from a full core (rated power is 1930 MWt) at ten days after the reactor is shutdown. This is less than 95% of the 5.5 MWt which JCP&L calculated for the total heat load. The difference, approximately .2 MWt, adequately accounts for the heat from the last two 1/4 cores to be unloaded. The JCP&L calculation of the heat load for the normal 1/4 core refueling case is also greater than would be obtained from use of NRC Technical Position APCSB 9-2. Thus, we find that JCP&L's calculations of the decay heat loads are adequately conservative.

At the present design heat removal rate of one MWt for a ΔT of 17°F the spent fuel pool cooling system will be capable of removing 2.06 MWt at the maximum heat sink temperature of 90°F while maintaining a 125°F spent fuel pool outlet temperature. This is adequate for the normal 1/4 core offload since the decay heat calculations show that the actual heat load for this case will be less than 1.85 MWt. However, for the full core offload at ten days after the reactor is shut down, a heat removal capability of about 5 MWt will be needed. Consequently, for this system to stay within the maximum 140°F spent fuel pool outlet temperature, a heat sink temperature of less than 50°F is required. If the heat sink temperature is greater than 50°F, retention of the core in the reactor vessel for a period in excess of ten days would be required for the full core offload case.

In order to minimize delays in unloading a full core JCP&L plans to install, prior to the core offload scheduled for April 1977, two new full capacity pumps and one heat exchanger in parallel with the two existing pumps and heat exchangers. The existing cross connect capability between the fuel pool cooling system and the "A" heat exchanger of the Shutdown Cooling System will be maintained. A review of existing systems by JCP&L revealed that with proper valve line-up, fuel pool water can be recirculated at 500 gpm through one main condenser to provide 8.9×10^6 BTU/hr additional cooling capacity. The new heat exchanger will be rated at 19 ± 1 10^6 BTU/hr (5·5MWt) which is sufficient to maintain pool temperature below 125°F when the core is in the pool and all of the remaining racks contain spent fuel assemblies. This modification is being designed in accordance with NRC's Standard Review Plan 9.1.3; that is, the new additional cooling system will be capable of withstanding the effects of the Safe Shutdown Earthquake and loss of offsite power coincident with single active component failure. On this basis the new system, we have concluded, is acceptable. The new system will be operated very infrequently, i.e., whenever the full core is unloaded. Surveillance will be accomplished, therefore, prior to each anticipated use to assure acceptable performance when placed into operation. A full core cannot be offloaded within ten days after reactor shutdown because of the time requirements to prepare for defueling. By this time reactor decay heat levels will be reduced to levels that are within the cooling capability limits of the new spent fuel pool cooling system. The potential for pool overheating is therefore acceptably low because of the improved reliability of the modified spent fuel pool cooling system.

We further conclude that (1) for the normal refueling case, with the existing spent fuel cooling system operating as designed, the temperature of the outlet water from the fuel pool will not exceed 125°F, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the use of this system in the proposed manner.

4. Fuel Handling and Installation of Racks

The Oyster Creek spent fuel pool is equipped with a cask drop protection system. This was found acceptable by the NRC in its evaluation of Amendment 68 to the FDSAR, and it has been used for some time in the shipment of fuel assemblies offsite.

Since there are irradiated fuel assemblies in the pool, the water cannot be drained to install the new racks. JCP&L states that the fuel assemblies that are in the pool will be removed to a remote area of the pool prior to bringing in a new rack, which will weigh less than 5200 pounds.

Since the stored fuel assemblies are protected by an approved cask drop protection system, the likelihood of a cask tip, drop, or swing accident wherein the fuel assembly spacing would be reduced to a more reactive geometry, i.e., a geometry where the neutron multiplication factor is increased, is considered to be extremely remote.

Moving fuel assemblies to a remote area of the pool prior to bringing in the new fuel storage racks, will eliminate the possibility of a rack drop, tip or swing accident that could cause a compression in the lattice geometry of stored fuel assemblies. Also, since the rack weighs less than 5200 pounds and will be under water when it is in the vicinity of any stored fuel assemblies there is additional assurance that a rack handling accident will not cause an increase in neutron multiplication factor in the fuel pool.

By using the same precautions that are used in handling the fuel cask when fuel is shipped offsite, installation of the modified spent fuel storage racks can be completed without jeopardizing the plant's cool down or spent fuel cooling capability.

We consider the licensee's cask drop protection system adequate for the prevention of cask tip accidents. The dashpot structure and fuel pool structure are adequate for loadings imposed during postulated cask tip accidents. The cask travel will be limited to the specific path and other heavy loads will not be carried over spent fuel. Movement of the 100 ton fuel cask assumed in the cask drop analyses will not be permitted until the details of the means used to limit the height to which the cask can be raised over the operating deck have been submitted by the licensee and approved by the NRC staff. The consequences of fuel handling accidents therefore remain unchanged from those presented in our SER dated December 1968.

We conclude that there is reasonable assurance that any postulated accident associated with the installation of the new racks will not cause the neutron multiplication factor in the fuel pool to exceed the NRC accepted value of 0.95 or jeopardize the plant's cool down or the spent fuel pool's cooling capability.

Conclusion

We have determined that the proposed modification to the spent fuel pool storage racks is acceptable because (1) the structural design is adequate, (2) the new storage racks will preclude criticality for the currently approved Oyster Creek fuel assemblies or fuel assemblies with even higher average ^{235}U enrichments that are less than 15.6 grams of ^{235}U per longitudinal centimeter of fuel assembly, (3) the spent fuel pool can be adequately cooled and (4) the modification will be completed without damage to stored fuel assemblies sufficient to cause criticality. We have therefore determined 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: March 30, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 22 to Provisional Operating License No. DPR-16 issued to Jersey Central Power & Light Company which revised Technical Specifications for operation of Oyster Creek Nuclear Generating Station, located in Ocean County, New Jersey. The amendment is effective as of its date of issuance.

The amendment will increase the spent fuel pool storage capacity from 840 to 1800 fuel assemblies. The increase will (1) provide storage for all spent fuel assemblies removed from the core between the present time and 1984, (2) provide sufficient additional fuel assembly storage capacity that the entire core (560 fuel assemblies) can be removed from the reactor vessel and stored in the spent fuel pool and (3) continue to accommodate one fuel assembly shipping cask for offsite shipping of spent fuel assemblies from the Oyster Creek spent fuel pool when offsite fuel shipment is resumed at some indefinite future date within the next 8 years.

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 Proposed Issuance of Amendment to Provisional Operating License in connection with this action was published in the FEDERAL REGISTER on April 22, 1976 (41 FR 16891). 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 for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because there will be no 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 Oyster Creek Nuclear Generating Station in December 1974 in the FEDERAL REGISTER.

For further details with respect to this action, see (1) the application for amendment dated March 18, 1976 and supplements dated August 11, 1976, November 30, 1976, January 18, 1977 and February 23, 1977, (2) Amendment No. 22 to License No. DPR-16, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Ocean County Library, Brick Township Branch, 401 Chambers Bridge Road, Brick Town, New Jersey 08723. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 30th day of March 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors