November 15, 1996

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-219-OLA
GPU NUCLEAR CORPORATION	)	(Tech. Spec. 5.3.1.B)
	)	
(Oyster Creek Nuclear Generating Station)	)	ASLBP No. 96-717-02-OLA

AFFIDAVIT OF JOHN C. FORNICOLA

County of Morris	)	
	)	ss.
State of New Jersey	)	

John C. Fornicola, being duly sworn according to law, appears and states as follows:

1. My name is John C. Fornicola. I am the Licensing and Regulatory Affairs Director for GPU Nuclear Corporation ("GPUN" or "Licensee"), responsible for licensing and regulatory affairs related to Oyster Creek Nuclear Generating Station ("Oyster Creek"). My business address is 1 Upper Pond Road, Parsippany, New Jersey, 07054. A summary of my professional qualifications is attached as Exhibit 1 hereto.

2. I am providing this affidavit in support of GPUN's Motion for Summary Disposition in this proceeding. I have knowledge of the statements provided below and believe them to be true and correct to the best knowledge, information, and belief. I am competent to testify to the matters stated herein.

3. On April 15, 1996, GPUN requested that Technical Specification 5.3.1.B be modified to add a second subpart as follows:

> 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.
>
> 2. The shield plug and associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

GPUN, Oyster Creek Technical Specification Change Request No. 244 (Apr. 15, 1996), attached as Exhibit 2 hereto. Pursuant to a final no significant hazards consideration determination, the NRC issued this amendment on November 7, 1996.

4. Prior to the November 7, 1996 amendment, Technical Specification 5.3.1.B of the operating license for Oyster Creek stated:

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

5. GPUN believes that Technical Specification 5.3.1.B for Oyster Creek has always applied specifically to spent fuel stored in storage racks and has never applied to the Cask Drop Protection System ("CDPS"). GPUN did not submit the Technical Specification change request to change the meaning of the Technical Specification as GPUN understands it, but rather to make its meaning more explicit. In light of the current regulatory climate in which both the NRC and licensees are particularly sensitive to the need for a well-defined and understood licensing basis, it was suggested that an amendment clarifying the technical specification would be desirable.

6. The original Technical Specification 5.3.1.B, as quoted in paragraph 4 above, was adopted over three years prior to the initial publication of NUREG-0612. It was not adopted in response to NUREG-0612. Rather, it was adopted for Oyster Creek as part of a 1977 amendment which increased the spent fuel pool storage capacity from 840 to 1800 fuel assemblies by replacing existing fuel storage racks with new closer spaced storage racks. U.S. Nuclear Regulatory Commission, Issuance of Amendment No. 22 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Mar. 30, 1977), attached as Exhibit 3 hereto. As discussed below, the correspondence related to this amendment shows that it was intended to apply only to spent fuel stored in the storage racks in the spent fuel pool.

7. On March, 18, 1976, GPUN applied for an amendment to expand the storage capacity of the spent fuel storage area by replacing the existing storage racks with new closer spaced storage racks. Jersey Central Power & Light Company, Request for Amendment to Provisional Operating License No. DPR-16 -- Technical Specification Change Request No. 44 and Facility Description and Safety Analysis Report ("FDSAR") Amendment No. 78 (Mar. 18, 1976), attached as Exhibit 4 hereto. The need for increased storage capacity was due, in part, to the shutdown of West Valley and the unavailability of reprocessing services. This amendment request specifically addressed the spent fuel storage racks and did not include the CDPS.

8. The CDPS was approved by the Atomic Energy Commission (the forerunner of the Nuclear Regulatory Commission ("NRC")) in 1973 and subsequently installed at Oyster Creek. The primary purpose of the CDPS is to protect the spent fuel pool structure from the potential drop of

a load during heavy load movements related to the packaging of spent fuel for transfer out of the facility. The CDPS spent fuel packaging area is both physically distinct and separate from the spent fuel storage area that is made up of storage racks for spent fuel.

9. The initial amendment request for the spent fuel pool expansion clearly differentiated the spent fuel storage racks area as separate and distinct from the CDPS. See Exhibit 4 (Initial Request for Amendment 22) at pages 10.0-16, 10.0-17 of the FDSAR amendment. This differentiation between the spent fuel storage area and the CDPS is continued throughout subsequent correspondence between the NRC staff and the Licensee on the amendment request.

10. Although Technical Specification 5.3.1.B was not part of the initial application, in the process of reviewing the amendment request the NRC staff asked questions about the structural integrity of the new spent fuel storage racks which eventually resulted in Technical Specification 5.3.1.B. U.S. Nuclear Regulatory Commission, Request for Additional Information, Oyster Creek Nuclear Generating Station Spent Fuel Pool - Increased Storage Capacity (June 24, 1976), attached as Exhibit 5 hereto, at 1-6.

11. Question 39 from the NRC staff asked the Licensee to:

> Provide (1) the number of bundles that could be struck by a cask fall or tip, including effects of any superstructure on the cask; (2) a conservative analysis of fission product release from fuel bundles potentially subject to impact assuming that the most recently off-loaded fuel is in the impact area; (3) a realistic (best estimate) radiological analysis of a cask fall or tip; and (4) any technical specifications proposed on the decay time required prior to loading storage positions within the zone which could be struck by a cask fall or tip.

Exhibit 5 at 5. Licensee response to Question 39 was that "(a) cask drop accident on or near stored fuel assemblies is not anticipated since the Oyster Creek spent fuel pool is equipped with a cask drop protection system (CDPS)," and the cask "will not be moved over the fuel storage area at any time." Jersey Central Power & Light Company, Supplement No. 1 to Facility Description and Safety Analysis Report Amendment No. 78 (Aug. 11, 1976), attached as Exhibit 6 hereto, at page 39-1 of the FDSAR amendment. This again clearly differentiates the CDPS from the location of "stored fuel assemblies" and the "fuel storage area."

12. Question 40 from the NRC staff asked the Licensee to:

> Discuss the overhead cask handling system from the points of view of (1) yoke and/or cable failure, and (2) braking devices, their capacity and effect on the ability of the handling system to withstand possible sudden decelerations induced by rapid braking following a loss of power to the system. Discuss all typical loads that may be carried near or over the spent fuel pool.

Exhibit 5 (June 24, 1976 Request for Additional Information) at 5-6. Licensee responded to Question 40 that "(s)ince the cask will not be moved over the fuel storage area, a yoke and/or cable failure is not expected to have any effect on stored assemblies." Exhibit 6 (Aug. 11, 1976 Supplement to Amendment Request) at page 40-1 of the FDSAR amendment. This again differentiates the CDPS and the safe load path followed by the cask from "fuel storage area" and "stored assemblies." It is important to note, as Licensee has previously identified, that movement of the shield plug is subject to the same constraints as movements of the cask. Exhibit 2 (Technical Specification Change Request No. 244) at unnumbered page 4.

13. Licensee's response to Question 40 also added that "[d]uring normal operation loads over the spent fuel pool will be limited to spent fuel assemblies, weighing approximately 700 lbs." Exhibit 6 (Aug. 11, 1976 Supplement to Amendment Request) at page 40-1 of FDSAR amendment.

14. As a follow-up, the NRC staff requested the Licensee to "[p]ropose Technical Specifications to limit the weight of loads moved over stored fuel in the spent fuel pool." Jersey Central Power & Light Company, Revision No. 1 to Technical Specification Change Request No. 44 and Addendum No. 1 to Supplement No. 1 Amendment No. 78 of the Facility Description and Safety Analysis Report (Nov. 30, 1976), attached hereto as Exhibit 7, at page 40-2 of the FDSAR amendment. The Licensee responded with "a proposed Technical Specification Change to limit the maximum weight of loads moved over the stored fuel in the spent fuel pool . . . ." Id. The proposed technical specification was labeled 5.3.1.D, but was otherwise identical to Technical Specification 5.3.1.B prior to the November 7, 1996 amendment, and read:

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

Exhibit 7 at unnumbered 7 of the introductory materials.

15. The NRC issued the amendment to replace the existing fuel storage racks with higher capacity spent fuel storage racks on March 30, 1977, with the technical specification on loads over stored fuel adopted as proposed. Exhibit 3 (Issuance of Amendment 22) at unnumbered page 1 of the introductory materials.

16. The NRC staff request, Licensee response, and new technical specification all include storage-related terms including "stored fuel," "stored irradiated fuel," and "fuel storage facility." This usage is consistent with the terms discussed earlier, including "fuel storage area" and "stored assemblies," that are used throughout the documentation of the amendment request.

17. In issuing the amendment to change the spent fuel storage racks, the NRC specifically stated that the amendment will:

> continue to accommodate one fuel assembly shipping cask for off-site shipping of spent fuel assemblies from the Oyster Creek spent fuel pool when offsite spent fuel shipment is resumed at some indefinite future date.

Exhibit 3 (Issuance of Amendment 22) at unnumbered 1 of the introductory materials. This NRC statement shows that both the staff and the Licensee understood that adopting Technical Specification 5.3.1.B, which is part of the subject amendment, does not prohibit future "shipping of spent fuel assemblies from the Oyster Creek spent fuel pool." The NRC statement also indicates that the packaging area for loading shipping casks is separate and distinct from the spent fuel storage area. This is consistent with the questions and responses related to Amendment 22, discussed above, that differentiate the CDPS from the location of "stored fuel assemblies" and the "fuel storage area."

18. Technical Specification 5.3.1.B has always allowed spent fuel to be packaged in the CDPS for transport out of the facility. The NRC staff and Licensee documentation related to the adoption of the original Technical Specification 5.3.1.B show it applied specifically to "stored

fuel" in the "fuel storage area" and it did not apply to the CDPS, where the shield plug would be handled.

19. The NRC subsequently published NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," in July 1980. NUREG-0612 identified a number of recommendations including Interim Protection measure (1) that recommended licensees implement a standard technical specification similar to Oyster Creek's Technical Specification 5.3.1.B. This recommendation was not adopted by the NRC in the Generic Letters following NUREG-0612. However, even if this recommendation had been adopted, it would not have altered the meaning or required any amendment or new interpretation of Technical Specification 5.3.1.B, because the interim technical specification recommended by NUREG-0612, the Standard Technical Specification on "Crane Travel - Spent Fuel Storage Pool," U.S. Nuclear Regulatory Commission, NUREG-0123, Standard Technical Specifications for General Electric Boiling Water Reactors, at 3/4 9-9 (Rev. 2 1979), attached hereto as Exhibit 8, was likewise limited to heavy loads over spent fuel stored in the storage racks.

20. This understanding of the meaning of Technical Specification 5.3.1.B is consistent with past practice. In 1975-1976 (prior to the amendment adding Technical Specification 5.3.1.B), GPUN packaged and transported 224 spent fuel assemblies from the Oyster Creek facility to the West Valley reprocessing plant. The NFS-4 cask was used to transport the spent fuel. The transportation cask was loaded in the CDPS and the packaging operation included moving the heavy shielded cask lid over spent fuel in the transportation cask within the CDPS.

21. In 1984-1985 (after the issuance of Technical Specification 5.3.1.B), the 224 spent fuel assemblies that had been shipped to West Valley were transported back to Oyster Creek. The TN-9 transportation cask was used to transport the spent fuel. The TN-9 cask was opened and unloaded following Oyster Creek Procedure No. A15C-45420. The transportation cask was opened in the CDPS and the unloading operation included moving the heavy shielded cask lid over spent fuel in the transportation cask within the CDPS. This was viewed as permissible under the technical specifications, because Technical Specification 5.3.1.B applied only to stored spent fuel.

22. This same situation occurred at the Consumers Power Company ("Consumers") Palisades Nuclear Plant in 1992. Palisades has a Technical Specification 3.21.2 that prohibits the movement of heavy loads "over fuel stored in the main pool zone." Consumers Power Co., Response to NRC Bulletin 96-02· Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment (May 16, 1996), attached as Exhibit 9 hereto, at page 2 of attachment (emphasis added). This technical specification is comparable to Oyster Creek Technical Specification 5.3.1.B because it specifically addresses heavy load movements over stored fuel. Palisades' dry storage activities required shield covers and other heavy loads to be moved over spent fuel in the transfer cask in the packaging area of the pool. Id. at pages 2-3 of attachment. Based on conversations between Consumers and the NRC staff, it was determined that spent fuel in the transfer cask was not subject to the requirements of Technical Specification

3.21.2 because the technical specification applied only to stored fuel, and fuel in the transfer cask is not being stored but is rather in transit. Id. at Page 3 of Attachment.

23. In summary, Technical Specification 5.3.1.B was originally issued before NUREG-0612, and it has always applied only to stored spent fuel in the fuel storage area, not to spent fuel in the CDPS being packaged for movement out of the reactor building. The current change is merely a clarification of this intended scope and meaning. Further, even if the technical specification were interpreted in light of the subsequent NUREG-0612 recommendations (including the interim recommendation that was not adopted in the generic letters), its scope and meaning would be unaltered. The interim technical specification recommended by NUREG-0612 also applies only to spent fuel stored in racks. Accordingly, there is no restriction on changing Technical Specification 5.3.1.B as GPUN requested.

John C Fornicola

Subscribed and sworn to before me
this 15th day of November, 1996.

Michele E. Shaffer

Notary Public

Michele E. Shaffer - Notary Public State of New Jersey

My commission expires 9-24-99.

J79893-41 / DOCSDC1

November 15, 1996

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-219-OLA
GPU NUCLEAR CORPORATION	)	(Tech. Spec. 5.3.1.B)
	)	
(Oyster Creek Nuclear Generating Station)	)	ASLBP No. 96-717-02-OLA

AFFIDAVIT OF JOHN C. FORNICOLA

EXHIBITS 1-9

377314-01 / DOCSDC1

Exhibit 1

JOHN C. FORNICOLA
SUMMARY OF PROFESSIONAL QUALIFICATIONS

Current Position - Licensing and Regulatory Affairs Director, GPU Nuclear Corporation

- Started in 1992.
- Responsible for the nuclear licensing and regulatory affairs activities for all GPU Nuclear sites.
- GPUN's principal interface with the NRC, PA and NJ State Agencies.
- GPUN's Administrative Point of Contact for the Institute of Nuclear Power Operations (INPO) and the Nuclear Energy Institute (NEI).

Previous Nuclear Experience

United States Navy

- Served aboard several nuclear submarines.
- Trained on overall operations of the plant and held several supervisory positions.

Quality Assurance Engineer, Metropolitan Edison Company

- Operations and maintenance related quality control surveillance activities.
- Nondestructive examination activities.
- Auditing and coordination of quality control inservice inspection.
- Outage planning and scheduling activities.

Quality Assurance Engineer, Allis Chalmers Nuclear Components Division

- Project responsibilities relating to the fabrication of various nuclear components.

Operations Quality Assurance Supervisor - GPU Nuclear Corporation

- Started in 1980; subsequently progressed through several managerial positions.

Three Mile Island Quality Assurance Manager - GPU Nuclear Corporation

- Started in 1986.
- Responsible for the implementation of the Quality Assurance Program on site, including Quality Control, Procurement Quality Assurance, and Operations Quality Assurance.

Director of Quality Assurance - GPU Nuclear Corporation

- Temporary assignment in 1992.
- Responsible for the implementation of the GPUN Quality Assurance program corporate wide.

Education

B.S. Degree (Mechanical Engineering) - Pennsylvania State University
Master's Degree (Administration) - Pennsylvania State University

8/94

Exhibit 2

Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number

April 15, 1996
6730-96-2087

U.S. Nuclear Regulatory Commission
Att: Document Control Desk
Washington D.C. 20555

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station (OCNGS)
Docket No. 219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

In accordance with 10 CFR 50.4(b)(1), enclosed is a Technical Specification Change Request (TSCR) No. 244. Also enclosed is a Certificate of Service for this request certifying service to the chief executive of the township in which the facility is located, as well as the designated official of the State of New Jersey Bureau of Nuclear Engineering.

The purpose of this TSCR is to revise specification 5.3.1.B of the Oyster Creek Technical Specifications. The current specification prohibits handling a load greater in weight than one fuel assembly over irradiated fuel in the spent fuel storage facility. The proposed change will facilitate the off load of spent fuel to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI). Specifically, the shield plug for the dry shielded canister (DSC) and the associated lifting hardware will be moved over irradiated fuel which is contained in the DSC within the transfer cask located in the Cask Drop Protection System (CDPS).

Pursuant to 10 CFR 50.91(a)(1), enclosed is an analysis applying the standards of 10 CFR 50.92 to make a determination of no significant hazards consideration.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachments
cc: Administrator, Region I
NRC Project Manager
NRC Resident Inspector

Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number

April 15, 1996
C321-96-2087

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, NJ 08625

Dear Mr. Tosch:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

Pursuant to 10 CFR 50.91(b)(1), please find enclosed a copy of the subject document which was filed with the United States Nuclear Regulatory Commission on April 15, 1996.

Very truly yours,

Michael B. Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachment
DPK/plp

Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number

April 15, 1996
C321-96-2087

The Honorable John C. Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Dear Mayor Parker:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 244
Revise Specification 5.3.1 Concerning Handling Heavy Loads over Irradiated Fuel

Enclosed herewith is one copy of Technical Specification Change Request No. 244, for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the United States Nuclear Regulatory Commission on April 15, 1996.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President & Director
Oyster Creek

Attachment

DPK/gl

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
GPU Nuclear Corporation)
Docket No. 50-219)

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 244, for Oyster Creek Nuclear Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on April 15, 1996, has this day of April 15, 1996, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable John Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

BY Michael B. Roche
Michael B. Roche
Vice President and Director
Oyster Creek

GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Facility Operating
License No. DPR-16

Technical Specification Change Request No. 244
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 244, to the Oyster Creek Nuclear Generating Station Operating License, a change to pages 5.3-1 and 5.3-2.

BY Michael B Roche
Michael B. Roche
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 15th day of April 1996.

Geraldine E Levin
A Notary Public of NJ

GERALDINE E. LEVIN
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 02/08/2000

5.3 AUXILIARY EQUIPMENT

5.3.1 Fuel Storage

A. The fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties.

B. 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.

2. The shield plug and the associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

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

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

E. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 2645.

BASIS

The specification of a K-effective less than or equal to 0.95 in fuel storage facilities assures an ample margin from criticality. This limit applies to unirradiated fuel in both the dry storage vault and the spent fuel racks as well as irradiated fuel in the spent fuel racks. Criticality analyses were performed on the poison racks to ensure that a K-effective of 0.95 would not be exceeded. The analyses took credit for burnable poisons in the fuel and included manufacturing tolerances and uncertainties as described in Section 9.1 of the FSAR. Calculational uncertainties described in 5.3.1.A are explicitly defined in FSAR Section 9.1.2.3.9. Any fuel stored in the fuel storage facilities shall be bounded by the analyses in these reference documents.

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 (1,2,3) and that dropped waste cans will not damage the pool liner.

Administrative controls over crane movements, which include mechanical rail stops, serve to prevent travel of the crane outside the analyzed load path over the cask drop protection system. A safety factor greater than 10 with respect to ultimate strength, and redundant shield plug lift cables provide adequate margin for the shield plug lift. These features, combined with operator training and required inspections, contribute to the determination that dropping the shield plug onto a loaded dry shielded canister in the spent fuel pool is not a credible event.

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 (4) 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 less than or equal 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 the 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 temperatures 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 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 (5) (6). 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. The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F (5).

Since the cooled fuel pool water returns at the bottom of the pool and the heated water is removed from the surface, the average of the surface temperature and the fuel pool cooling return water is an appropriate estimate of the average bulk temperature; alternately the pool surface temperature could be conservatively used.

References

1. Amendment No. 78 to FDSAR (Section 7)
2. Supplement No. 1 to Amendment No. 78 to the FDSAR (Question 12)
3. Supplement No. 1 to Amendment 78 of the FDSAR (Question 40)
4. Supplement No. 1 to Amendment 68 of the FDSAR
5. Revision No. 1 to Addendum 2 to Supplement No. 1 to Amendment No. 78 of FDSAR (Questions 5 and 10)
6. FDSAR Amendment No. 79
7. Deleted

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) No. 244

GPU Nuclear requests the following replacement pages be inserted into existing Technical Specifications:

Replace existing pages 5.3-1 and 5.3-2 with the attached revised replacement pages 5.3-1 and 5.3-2.

II. REASON FOR CHANGE

The current specification 5.3.1.B requires that "Loads greater than weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility". This restriction is based upon the structural strength of the fuel racks in which the spent fuel is stored and the damage that would occur if the load were dropped. The process of transferring spent fuel assemblies to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI) includes placing a dry shielded canister (DSC) and a transfer cask in the cask drop protection system (CDPS). That movement does not handle a heavy load over irradiated fuel. The DSC is then loaded with spent fuel assemblies. Before the DSC and the transfer cask in which it is contained can be removed from the spent fuel pool, the DSC shield plug must be lowered into the CDPS and placed atop the DSC. The current specification prohibits this movement since the shield plug and the lifting yoke weigh more than one fuel assembly and the DSC contains irradiated fuel.

III. SAFETY EVALUATION JUSTIFYING CHANGE

GPU Nuclear has evaluated the process of transferring spent fuel assemblies from the spent fuel pool to the ISFSI. That evaluation considers the safe load paths, the design features of the reactor building crane and the requirements of NUREG 0612.

The CDPS has been designed to mitigate a cask drop into the spent fuel pool. The transfer path for the cask centerline is on a controlled path width of six inches in the north-south and east-west directions. Visual aids are used to control the motion of the cask centerline to the prescribed transfer path. Mechanical rail stops are installed to prevent travel of the crane outside the analyzed load path over the cask drop protection system. Stops are installed for limiting bridge movements in the north-south direction and for limiting trolley movements in the west direction. The movement of the shield plug would be in accordance with these same constraints. The weight of the load, however, would be considerably different. The shield plug weighs approximately 8,000 pounds and the lifting yoke weighs about 6,200 pounds.

A series of modifications have been made to enhance the crane's performance and reliability by improving the instrumentation and controls. These modifications include:

- Various crane monitoring systems have been installed. These include drum over-speed detection, mechanical drive train continuity detection, wire rope spooling monitor, fault display and reset panel and hoist speed indication.

- Phase loss/phase reversal protection has been installed. Phase loss results in substantial loss of drive motor torque and possible load drop.

- A power circuit upper limit switch to directly interrupt power to the hoist motor was installed. This reduces the possibility of two-blocking as a result of failure of existing control circuit limit switches.

- A load cell weight display was installed in the cab to provide an indication for load hang-ups and over-capacity lifts.

- The magnetic drive controllers were replaced. The new variable frequency drive (VFD) controllers provide smooth and precise speed control along with torque limitation, reducing the possibility of a load snatch.

- New controls were installed in the cab that provide spring control to normal function. These controls considered human factors in their design.

The reactor building (RB) crane has a main hoist capacity of 100 tons. The actual safety factors of the main crane for its 100 ton rated load are: cables 6.5:1; main hoist gearing 5.2:1; and main hoist brakes 120% capacity. As a result, when moving the shield plug and the lifting yoke with a combined weight of approximately 7 tons, a safety factor greater than 14 will be provided, based on the RB crane 100 ton rated capacity. For the lifting yoke, a safety factor greater than 26 will be provided, based on the lifting yoke 105 ton rated capacity. The least conservative safety factor is that for the wire rope assemblies. That safety factor is 11:1, based on the ultimate load of 22,800 lbs. Furthermore, the wire rope assemblies are redundant and each of the four has sufficient capacity to support the total weight of the shield plug.

In addition, GPU Nuclear has developed an error free plan for the movement of spent fuel assemblies to the ISFSI. That plan includes a dedicated management team and a dedicated crew who will be trained and on shift. Detailed operating instructions/procedures will be developed and mock-up training and a dry run will be conducted. A special crane inspection will be performed prior to each dry fuel storage campaign. The main hoist coupling, shafts, and hook will be examined by NDE prior to each campaign. Plant procedures for the reactor building crane satisfy the inspection, testing and maintenance criteria of ANSI B30.2.

The design features and modifications to the reactor building crane increase its reliability and enhance its performance. The safety factors of the reactor building crane relative to this load exceed 10 to 1. Personnel training, and crane inspections, testing, and maintenance will be in accordance with ANSI B30.2. Therefore, dropping the DSC shield plug onto a loaded DSC in the spent fuel pool is not considered a credible event.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this TSCR poses no significant hazard as defined by 10 CFR 50.92.

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

The design features and capacity of the reactor building crane provide a significant safety factor. In addition, personnel training and other administrative controls further reduce risk. Thus, the dropping of the DSC shield plug onto a loaded DSC and causing damage to the spent fuel assemblies is not a credible event. Therefore, it does not increase the probability of or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR.

This activity will not create the possibility of a new or different type of accident than previously evaluated in the SAR because the proposed heavy load handling exception does not create a new credible accident scenario. Dropping the shield plug on a loaded DSC and damaging spent fuel assemblies is not considered a credible event.

3. State the basis for the determination that the margin of safety is not reduced.

This activity will not involve a significant reduction in the margin of safety because the proposed heavy load handling evolution does not create a credible accident scenario.

V. IMPLEMENTATION

GPUN requests that the amendment authorizing this change be effective upon issuance.

Exhibit 3

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

March 30, 1977

Docket No. 50-219 - 2035

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,

George Lear

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

Enclosures and ccs:
See next page

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

cc:

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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 umho/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. Spent Fuel Cooling

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

Docket - 50-219 - - 329

Jersey Central Power & Light Company

Exhibit 4

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

Member of the General Public Utilities Corporation System

March 18, 1976
GD-76-013

Mr. Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Stello:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Request for Amendment to Provisional Operating License No. DPR-16 -- Technical Specification Change Request No. 44 and Facility Description and Safety Analysis Report Amendment No. 78

Pursuant to Title 10, Code of Federal Regulations, Section 50.59, three signed originals and thirty-seven conformed copies of Jersey Central Power & Light Company's request for Amendment to Appendix A of the Oyster Creek Nuclear Generating Station's Provisional Operating License No. DPR-16 and amendment to the Facility Description and Safety Analysis Report are herein submitted. This request incorporates Technical Specification Change Request No. 44 and Amendment No. 78.

These proposed Amendments incorporate the modifications and specifications necessary to accommodate the planned increase in storage capacity of the spent fuel storage pool of the Oyster Creek Station, which is necessary to ameliorate the shortage of spent fuel storage capacity. The planned modification and evaluation are delineated in the attached Amendment No. 78 to the FDSAR. Note that no Amendment to the Operating License is included because of our proposed Operating License Amendment of December 18, 1975. This December 18, 1975, proposal was initiated by Mr. Lear's request of December 16, 1974, for changes to facilitate the licensing of special nuclear, byproduct, and source materials. The change proposed for Paragraph 2.B.1 in the December 18, 1975, submittal limits the amount of special nuclear material received, possessed, or used by reference to the Facility Description and Safety Analysis Report. We, therefore, request that you issue the proposed license Amendment of December 18, 1975, concomitantly with the attached FDSAR Amendment No. 78 and Technical Specification Change Request No. 44 and that the license Amendment reference Amendment No. 78 in Paragraph 2.B.1.

2814

The Plant Operations Review Committee (PORC) and the General Office Review Board (GORB) have reviewed these proposals. They have determined that one unreviewed safety question exists because of the change in the current Technical Specification limit for the multiplication factor for the spent fuel pool from 0.90 to 0.95. They have also determined that this proposed Technical Specification change and plant modification will not adversely affect plant safety.

Very truly yours,

Ivan R. Finfrock

Ivan R. Finfrock, Jr.
Vice President

pk

Enclosures

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Request for Amendment

Facility Description and Safety Analysis Report
Amendment No. 78

and

Technical Specification
Change Request No. 44
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 44 to the Oyster Creek Nuclear Generating Station Technical Specifications, modifications necessary to change the maximum effective neutron multiplication factor of the spent fuel storage pool from 0.90 to 0.95.

Included also is Amendment No. 78 to the Oyster Creek Nuclear Generating Station's Facility Description and Safety Analysis Report. This Amendment describes the planned spent fuel storage pool modification and presents the safety evaluation of the modification.

JERSEY CENTRAL POWER & LIGHT COMPANY

By ______________________
Ivan R. Finfrock, Jr.
Vice President

STATE OF NEW JERSEY)

COUNTY OF MORRIS)

Sworn and subscribed to be[fore] me this [illegible] day of [illegible], 1976.

____
Notary Public

PHYLLIS A. KABIS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires [illegible]

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N.J. 07960 • 201-539-6111

March 18, 1976
[illegible]-76-11

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Scanlon:

Enclosed is one copy of Technical Specification Change Request No. 44 for the Technical Specifications and one copy of Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station.

This Technical Specification Change Request and Amendment to the Facility Description and Safety Analysis Report were filed with the United States Nuclear Regulatory Commission on March 18, 1976.

Very truly yours,

Ivan R. Finfrock, Jr.
Vice President

pk

Enclosures

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY)

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 44 for the Technical Specifications and a copy of Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station, both dated March 18, 1976, and filed with the United States Nuclear Regulatory Commission on March 18, 1976, has this 18th day of March, 1976, been served on the Mayor of Lacey Township, Ocean County, New Jersey, by deposit in the United States mail, addressed as follows:

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

By Ivan R. Finfrock, Jr.
Ivan R. Finfrock, Jr.
Vice President

Dated: March 18, 1976

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION
(Docket No. 50-219)

PROVISIONAL OPERATING LICENSE NO. DPR-16

Applicant hereby requests the Commission to change Appendix A to the above-captioned license as follows:

1. Section to be Changed

   Section 5.3.1

2. Extent of Changes

   To change the maximum k_{eff} of the spent fuel storage facility from 0.90 to 0.95.

3. Changes Requested

   Replace Page 5.3-1 with the attached Page 5.3-1

4. Discussion

   Amendment No. 78, dated March 18, 1976 of the Oyster Creek Nuclear Generating Station's Facility Description and Safety Analysis Report describes the modification of the spent fuel storage pool to expand its capacity by replacing existing storage racks with compacted storage racks. Section 3 of the Amendment describes the criticality analysis. It shows that even with conservative assumptions and allowance for tolerances, temperature effects, void effects, calculational uncertainties, etc., the k_{∞} of the Oyster Creek spent fuel pool is well below 0.95. The k_{eff} 0.095 specification provides ample margin from criticality and is consistent with ANSI N18.2-1973 and the Commission's Standard Technical Specifications for Boiling Water Reactor Plants, dated September 1, 1975. Note that reactors are only required to maintain .01 Δk margin from criticality.

5.3 AUXILIARY EQUIPMENT

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_{eff} less than or equal to 0.95.

Jersey Central Power & Light Company

OYSTER CREEK
NUCLEAR GENERATING STATION

FDSAR AMENDMENT 78

SPENT FUEL POOL EXPANSION

OYSTER CREEK NUCLEAR GENERATING STATION

SPENT FUEL POOL EXPANSION

FDSAR

AMENDMENT NO. 78

March 1976

TABLE OF CONTENTS

1.0 INTRODUCTION

The spent fuel pool at the Oyster Creek Nuclear Generating Station has capacity at present for 840 fuel assemblies and contained 526 assemblies as of January 1976. Spent fuel shipping has recently removed 224 assemblies from the Oyster Creek site, but there is no contractual agreement to ship other spent fuel. This means that after the present refueling (December 1975--February 1976) the Oyster Creek facility will not be able to offload a full core of 560 assemblies. To ameliorate this shortage of spent fuel storage capacity, the Oyster Creek Station is planning to increase the spent fuel pool capacity by replacing the current spent fuel racks, which have a 20 assembly storage capacity, with compacted assembly storage capacity. New 28-element racks will occupy the same space envelope as the present 20-element racks. Other compacted storage racks will be added in some of the areas now vacant in the spent fuel pool. These will have 28 or 32 assembly storage capacity. The 32-element racks will have the same spacing as the 28-element rack.

This increases the capacity from 840 to 1800 spent fuel assemblies and provides the capability to store all spent fuel assemblies through 1983 and still retain the capability to accommodate one shipping cask and removal of the complete core from the vessel. During refueling periods and whenever the shipping cask is not in the pool, the cask area will be available to perform underwater inspection of, or underwater mechanical operations on, radioactive components.

This Amendment to the FDSAR provides a safety evaluation and description of the proposed rack design and installation. The pool capacity expansion will begin as soon as possible.

2.0 DESIGN DESCRIPTION

2.1 Spent Fuel Racks

The new fuel rack is a modular design with fuel arranged in slabs, as shown in Figure 1.

The rack design meets the applicable parts of the codes and standards compiled in Table II-1 as the following discussion will show. All material used in the rack is ASTM A240, Type 304 stainless steel. The fuel cell regions of the rack are made up of rectangular stainless steel boxes with a base plate at the bottom of each box with two holes to support and position the fuel assemblies.

The flux trap region between fuel boxes is formed by boxes of 3-inch width. There are upper end caps that fit over the top of these boxes and are configured to provide "lead-in" for insertion of fuel assemblies into the adjacent fuel boxes (Figure 2). The box array is joined by welding to form a solid honeycomb structure of the pattern shown in Figure 1. In order to provide increased strength to meet seismic loads without adding lateral restraints, these compacted racks are joined mechanically in minimum groups of 8 (i.e., 2 x 4) as shown in Figure 3. These sets of racks are bolted to the support beams that are anchored to the bottom of the pool. All welding will be done in accordance with AWS Specification No. D.1.1-Rev. 2-74, Structural Welding Code, and the appropriate parts of the ASME Boiler and Pressure Vessel Code as indicated in Table II-1.

The new rack design does not impair the transfer of fuel assemblies nor does it prohibit the storage of reactor internals which must be removed for refueling. The Design Evaluation sections of this Amendment (Sections 2.0, 3.0, and 4.0) delineate the rack design as they describe the seismic, criticality, etc., capabilities of the racks.

2.2 Rack Base

Stainless steel "I" beams will be installed in the pool to provide a mounting base for the new racks (Figure 4). These beams are provided with leveling pads and lugs for the swing bolts that fasten them to the bottom of the pool. When the beams are leveled, they are firmly fastened to the bottom of the pool by existing swing bolts. Individual racks will be lowered onto the beams and bolted to them. The use of these beams provides flexibility in positioning racks and allows racks to be placed into areas where there are no swing bolts, thus increasing the total number of racks.

TABLE II-1

CODES, STANDARDS, AND REGULATORY REQUIREMENTS
APPLICABLE TO THE DESIGN AND FABRICATION OF THE REPLACEMENT
SPENT FUEL RACKS AS REFERENCED HEREIN

ANSI N 18.2 - 1973

NRC Regulatory Guides:
- 1.13 - Safety Guide No. 13 - March 10, 1971
- 1.26 - Rev. 2, 1975
- 1.29 - Rev. 1, 1973
- 1.92 - 1974

NRC Reg. 10CFR50, Appendix B, 1975

ASME Boiler & Pressure Vessel Code, Section IX, 1974 and latest Addenda

AISC Manual of Steel Construction, Part 5, 7th Edition, 1970

AWS Spec. No. D.1.1-Rev. 2 - 74, Structural Welding Code

Oyster Creek Facility Description and Safety Analysis Report

Oyster Creek Quality Assurance Plan

3.0 CRITICALITY ANALYSIS

The following discussion shows that the design and installation of the new spent fuel racks meet the applicable criteria of the standards and codes of Table II-1 for criticality design.

3.1 Analytical Technique

The LEOPARD[1] computer program was used to generate macroscopic cross sections for input to four energy group diffusion theory calculations which are performed with the PDQ-7[2] program. LEOPARD calculates the neutron energy spectrum over the entire energy range from thermal up to 10 Mev and determines averaged cross sections over appropriate energy groups. The fundamental methods used in the LEOPARD program are those used in the MUFT[3] and SOFOCATE[4] programs which were developed under the Naval Reactor Program and thus are well founded and extensively tested analytic techniques. In addition, Westinghouse Electric Corporation, the developers of the original LEOPARD program, demonstrated the accuracy of these methods by extensive analysis of measured critical assemblies consisting of slightly enriched UO_2 fuel rods[5].

In addition, Pickard, Lowe and Garrick, Inc. (PLG) has made a number of improvements to the LEOPARD program to increase its accuracy for the calculation of reactivities in systems which contain significant amounts of plutonium mixed with UO_2. PLG has tested the accuracy of these modifications by analyzing a series of UO_2 and PuO_2-UO_2 critical experiments. These benchmarking analyses not only demonstrate the improvements obtained for the analysis of PuO_2-UO_2 systems but also demonstrate that these modifications have not affected the accuracy of the PLG-modified LEOPARD program for calculations of slightly enriched UO_2 systems.

The UO_2 critical experiments chosen for benchmarking include variations in H_2O/UO_2 volume ratios, U-235 enrichments, pellet diameters and cladding materials. Although the LEOPARD model also accurately calculates the reactivity effects of soluble boron, these experiments have not been included in the benchmarking criticals since the spent fuel pool calculations do not involve soluble boron.

Neutron leakage was represented by using measured buckling input to infinite lattice LEOPARD calculations to represent the critical assembly. A summary of the LEOPARD results is shown in Table III-1 for the 27 measured criticals chosen as being directly applicable for benchmarking the model for spent fuel pool calculations. The average calculated k_{eff} is 0.9979 and the standard deviation from this average value is 0.0080 Δk. Reference 5 raised questions concerning the accuracy of the measured bucklings reported for

the experiments number 12 through 19. If these data are excluded, the average calculated k_{eff} for the remaining 19 experiments is 1.0006 with a standard deviation from this value of 0.0063 Δk.

The PDQ series of programs have been extensively developed and tested over a period of 20 years and there is no question that the current version, PDQ-7, is an accurate and reliable model for calculating the subcritical margin of the proposed spent fuel pool arrangement. The code or a mathematically equivalent method is used by all the U.S. suppliers of light water reactor cores and reload fuel. In addition, this code has received extensive utilization in the U.S. Naval Reactor Program where accuracy is of utmost importance.

As a specific demonstration of the accuracy of the calculational model used for the spent fuel pool calculations, the combined LEOPARD/PDQ-7 model has been used to calculate seven measured just critical assemblies. The criticals are high neutron leakage systems with a large variation in U/H_2O volume ratio and include parameters in the same range as those applicable to the proposed spent fuel pool design. Experiments including soluble boron are included in this demonstration since we are primarily interested in the ability of PDQ-7 to calculate neutron leakage effects. The use of soluble boron allows changes in the neutron leakage of the assembly while maintaining a uniform lattice and thus allows a better test of the accuracy of the model.

These LEOPARD/PDQ-7 calculations, shown in Table III-2 result in a calculated average k_{eff} of 0.9922 with a standard deviation about this value of 0.0014 Δk. These results, together with the previously discussed LEOPARD results, demonstrate that the proposed LEOPARD/PDQ-7 calculational model can calculate the reactivity of the proposed spent fuel pool arrangement with an accuracy of better than ± 0.01 Δk.

The PDQ-7 program is used in the final predictions of the reactivity of the spent fuel storage pool. The calculations are performed in four energy groups and take into account all the significant geometric details of the fuel bundles, fuel boxes, and major structural components. The geometry used for most of the calculations is a basic cell representing one-quarter of the area of a repeating array of two different sized stainless steel boxes, the larger of which will accommodate two fuel bundles. The specific geometry and dimensions of this basic cell are shown in Figure 5. Except for the space between rack modular assemblies, this basic cell is an excellent representation of the exact geometry of the spent fuel racks. The specific reactivity effects of the space between rack modular assemblies, as well as the reactivity effects at the ends of the rack modular assemblies, are evaluated with a detailed one-dimensional PDQ-7 model using flux weighted cross sections from the basic cell.

This one-dimensional model is shown to give results that are virtually identical to those obtained with the two-dimensional basic cell geometry.

Thus, the calculational approach is to use the basic cell to calculate the reactivity of an infinite array of uniform spent fuel racks and to account for any deviations of the actual spent fuel rack array from this assumed infinite array as pertubations on the calculated reactivity of the basic cell. The effects of mechanical tolerances are also treated as pertubations on the calculated reactivity of the basic cell. The fuel bundles were assumed to be unirradiated with a U-235 enrichment of 3.0 w/o, which is higher than any anticipated reload enrichment for Oyster Creek. The basic calculations also assume the fuel contains no burnable poison even though it would not be possible to utilize such fuel bundles in the Oyster Creek reactor and maintain an adequate shutdown margin. The effects of burnable poison on the actual maximum reactivity of a fuel bundle are considered as a pertubation on the basic reactivity calculations along with the reactivity effects of uncertainties and other inherent conservatisms in the basic cell calculations. Most of the calculations were performed at a uniform pool temperature of 80°F, but the reactivity effects of pool temperature are also taken into account as a pertubation on the basic cell calculations.

3.2 Tolerance Considerations

With nominal dimensions on the structural components, the calculated k_∞ of the basic cell at 80°F is 0.8980. With the worst combination of mechanical tolerances on the structural components, the k_∞ of the basic cell is 0.8991, which results in a pertubation of +0.0011 Δk. With the fuel bundles located in their most reactive positions, the k_∞ of the basic cell is $\sim$ 0.9023 which results in a pertubation of +$\sim$ .0043 Δk. Thus, the total pertubation on the basic cell reactivity due to mechanical tolerances and positioning uncertainties is +$\sim$ .0054 Δk. The stainless steel fuel and water boxes are nominally 0.090 inch thick with a tolerance of $\pm$ .004 inch. Assuming a worst case in which all boxes were at the minimum thickness of 0.086 inch, the k_∞ of the basic cell is 0.9006. Therefore, the maximum pertubation on the reactivity of the basic cell due to variations in stainless steel box thickness is 0.0026.

Most of the calculations with the basic cell geometry utilized a 28 x 27 two-dimensional array of mesh points. To test the adequacy of this mesh description, a calculation was run with a 56 x 54 mesh size and the resulting k_∞ was 0.8972. Thus, the pertubation on the basic cell due to mesh spacing effects is -0.008 Δk.

The k_∞ of the basic cell as a function of temperature is shown in Figure 6. With a maximum pool temperature of 200°F under the worst possible conditions the k_∞ is 0.9090, which results in a pertubation due to temperature effects of +0.0110 Δk. Although the overall steady-state reactivity temperature coefficient of the spent fuel pool is positive, the temperature coefficient of the fuel bundles is negative.

Voids within the fuel bundle strongly reduce k_∞. Calculations show that k_∞ is reduced by ~ 0.0073 for an increase from 0 to 4% voids with a uniform moderator and fuel temperature of 200°F. Calculations for voids in the bundle and in the water region surrounding the fuel bundle, but inside the stainless steel cell, are virtually identical to the in-bundle calculation.

A detailed one-dimensional model was used to calculate the reactivity effect of the spacing between modular fuel rack assemblies. The accuracy of this model was checked by running a one-dimensional representation of the basic cell using flux weighted cross sections from the two-dimensional basic cell calculation. This one-dimensional model is shown in Figure 7. The k_∞ calculated with the one-dimensional representation is 0.8972, which is in excellent agreement with the basic cell value of 0.8980. With minimum spacing between modular rack assemblies, the calculated k_∞ is 0.9004, which when compared to the infinite array k_∞ of 0.8972, results in a pertubation of 0.0032 Δk. A similar one-dimensional model was used to evaluate the reactivity effect of the opposite end of the modular rack assemblies where there is no stainless steel box other than the fuel box and a water reflector. This calculation results in a lower k_∞ than that of an infinite array and thus the pertubation is a negative one.

Since this negative pertubation cannot be easily quantified for application to the total array of racks in the pool, no credit will be assumed for this negative pertubation.

The basic cell was also used to evaluate the reactivity effect of axial neutron leakage. Using an axial buckling based on a 12-foot active fuel length with a total reflector savings of 15 cm, the calculated k_∞ of the basic cell is 0.8960. Thus, the reactivity pertubation due to axial neutron leakage is -0.0020 Δk.

A summary of the pertubations to the basic cell reactivity calculation is shown in Table III-3. Thus, the calculated reactivity of the spent fuel pool with ~ 1800 unirradiated bundles with 3.0 w/o U-235 and no burnable poison is 0.9174 for a pool temperature of 200°F.

Criterion 5.7.4.1 of ANSI N 18.2-1973 specifies that the effective multiplication factor of the spent fuel pool will not exceed 0.95 with new fuel of the highest anticipated enrichment in place. This

specification results in a required subcritical margin of 0.05 Δk. Since reactors are required to maintain a subcritical margin of only 0.01 Δk or less, the intent of the larger subcritical margin requirement for spent fuel pools must be to some extent to account for uncertainties in the evaluation of the multiplication factor.

3.3 Uncertainty Considerations

Although we firmly believe this was the intent of ANSI N 18.2, we further understand that the NRC Staff has concluded that the effective multiplication factor of the spent fuel must not exceed 0.95 including allowances for all uncertainties. We will, therefore, show that the propo[illegible] spent fuel pool design is compatible with the NRC Staff's interpretation of ANSI N 18.2.

In the foregoing discussion, we demonstrated that the uncertainty in the calculated k_{eff} with the model utilized for criticality calculations was less than 0.01 Δk. There are a number of conservatisms in the model's representation of the spent fuel pool, and these conservatisms more than compensate for the uncertainty in the calculational model.

The basic cell calculations of k_∞ apply to an infinite array of racks containing unirradiated fuel bundles with no burnable poisons and no net radial neutron leakage. The maximum reload batch size anticipated for Oyster Creek is less than 150 bundles. Therefore, even if the entire core were to be discharged shortly after the start of a fuel cycle, there would be at most 150 irradiated fuel bundles in the spent fuel pool. In such a situation, there would be significant neutron leakage from the 150 unirradiated bundles to surrounding irradiated bundles or to empty fuel locations or to the water reflector. It is conservatively estimated that the resulting radial neutron leakage would reduce the calculated reactivity of the basic cell by 0.0067 Δk.

The spacer grids utilized in the design of all current BWR fuel bundles contain inconel springs which result in parasitic neutron absorption which is not included in the basic cell calculations. The spacer grids are calculated to reduce the k_∞ of the basic cell by at least 0.002 Δk.

As discussed previously, the basic cell calculations make the conservative assumption that all fuel bundles are unirradiated and contain no burnable poisons. Calculations, verified by reactor operation, show that with the required burnable poison loadings, the maximum possible bundle k_∞ is at least 0.030 Δk less than the initial k_∞ of the bundle with no burnable poison. This effect would reduce the calculated k_∞ of the basic cell by at least 0.0300 Δk.

The inherent conservatisms in the analytical model are such as to reduce the calculated k_∞ of the basic cell by at least 0.0447 Δk (i.e., .0067 + .002 + .0360). This reduction in k_∞ is more than four times the possible increase in the k_∞ of the basic cell due to uncertainties in the analytical model. Therefore, the multiplication factor of the spent fuel pool is less than 0.883 (i.e., .9174 + .01 - .0447 = .8827) at the 1.0 σ level. Thus, the 95% bound is 0.893 assuming the normal approximation. The reactivity effect of a fuel channel around the fuel assembly is only +.0004 and is, therefore, negligible.

These analyses only take credit for the inherent neutron absorbing properties of the Type-304 stainless steel boxes which are the principal structural components of the spent fuel racks. Fe, Ni, Cr, and Mn account for 99% of the composition of Type-304 stainless steel and these are the only constituents which are considered to absorb neutron in these analyses. Other constituents, including impurities, will result in some small additional neutron absorption and further increase the subcriticality of the rack.

The fuel racks are designed to prevent a dropped fuel bundle from penetrating and occupying a position other than a normal fuel storage location. A dropped fuel bundle will probably end up in a final position that is somewhere between vertical and horizontal on top of the racks. The only positive effect of such a bundle on the reactivity of the rack would be by virtue of a reduction in axial neutron leakage from the rack. Since the calculations reported here show the total axial neutron leakage effect to be only 0.0020 Δk, a dropped fuel bundle would not have any significant effect on the reported maximum possible reactivity of the spent fuel storage rack. The reactivity effect of a fuel bundle on the side of a rack has been calculated to be +0.0127. The worst case tolerance, temperature, etc. k_∞ is still only 0.906 Δk.

This demonstrates that even the inclusion of extreme conservatisms does not violate the 0.95 Δk criteria of ANSI N 18.2-1973. Because of the well founded, conservative technique used for determination of the infinite multiplication factor, there is reasonable assurance that this spent fuel rack design will not cause undue risk to the public health and safety resulting from criticality considerations.

TABLE III-1

Case[+] Number	Reference Number	Enrichment (atom %)	H_2O/U Volume	Fuel Density (g/cm^3)	Pellet Diameter (cm)	Clad Diameter (cm)	Clad Thickness (cm)	Lattice Pitch (cm)	Critical Buckling m^{-2}	Calculated k_{eff}
1	11	2.734	2.18	10.18	0.7620	0.8594	0.04085	1.0287	40.75	1.0015
2	11	2.734	2.93	10.18	0.7620	0.8594	0.04085	1.1049	53.23	1.0052
3	11	2.734	3.[illegible]6	10.18	0.7620	0.8594	0.04085	1.1938	63.26	1.0043
4	12	2.734	7.02	10.18	0.7620	0.8594	0.04085	1.4554	65.64	1.0098
5	12	2.734	8.49	10.18	0.7620	0.8594	0.04085	1.5621	60.07	1.0118
6	12	2.734	10.38	10.18	0.7620	0.8594	0.04085	1.6891	52.92	1.0072
7	13	2.734	2.50	10.18	0.7620	0.8594	0.04085	1.0[illegible]	[illegible]7.5	1.0008
8	13	2.734	4.51	10.18	0.7620	0.8594	0.04085	[illegible]2	68.8	0.9987
9	13	3.745	2.50	10.37	0.7544	0.8600	0.0406	1.0617	68.3	1.0010
10	13	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.1	1.0025
11	14	3.745	4.51	10.37	0.7544	0.8600	0.0406	1.2522	95.68	1.0009
12	15	4.069	2.55	9.46	1.1278	1.2090	0.0406	1.5113	88.0	0.9859
13	15	4.069	2.14	9.46	1.1278	1.2090	0.0406	1.450	79.0	0.9830
14	16	4.069	2.59	9.45	1.1268	1.2701	0.07163	1.555	69.25	0.9999
15	16	4.069	3.53	9.45	1.1268	1.2701	0.07163	1.684	85.52	0.9958
16	16	4.069	8.02	9.45	1.1268	1.2701	0.07163	2.198	92.84	1.0040
17	16	4.069	9.90	9.45	1.1268	1.2701	0.07163	2.381	91.79	0.9872
18	16	3.037	2.64	9.28	1.1268	1.2701	0.07163	1.555	50.75	0.9916
19	16	3.037	8.16	9.28	1.1268	1.2701	0.07163	2.198	68.81	0.9809
20	8	0.714*	1.68	9.52	0.8570	0.9931	0.0592	1.3208	108.8	0.9912
21	8	0.714*	2.17	9.52	0.8570	0.9931	0.0592	1.4224	121.5	1.0029
22	8	0.714*	4.70	9.52	0.8570	0.9931	0.0592	1.8669	159.6	0.9944
23	8	0.714*	10.76	9.52	0.8570	0.9931	0.0592	2.6416	128.4	1.0008
24	9	0.729*	1.11	9.35	1.2827	1.4427	0.0800	1.7526	69.1	0.9902
25	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	104.72	1.0055
26	9	0.729*	3.49	9.35	1.2827	1.4427	0.0800	2.4785	79.5	0.9948
27	9	0.729*	1.54	9.35	1.2827	1.4427	0.0800	1.9050	90.0	0.9878

* These are PuO_2 in Natural UO_2 in Zr.

\+ Cases 1 through 19 are with stainless s[illegible] clad, Cases 20 through 27 are zircalloy clad.

TABLE III-2

WESTINGHOUSE UO_2 CRITICAL EXPERIMENTS

(References 6 and 7)

Expt	Boron (ppm)	H_2O/UO_2 (Volume)	Pitch (In)	k_{eff} (PDQ-7)
1	0	1.49	.600	.9905
2	0	2.42	.690	.9949
3	0	4.35	.848	.9921
4	0	6.21	.976	.9918
5	306.0	1.49	.600	.9912
6	536.4	1.49	.600	.9925
7	727.7	1.49	.600	.9926

TABLE III-3

Description of Reactivity Pertubation	Reactivity Effect, Δk
Mechanical spacing tolerance on stainless steel boxes and fuel position uncertainties	+0.0054
Mechanical tolerance on stainless steel box walls	+0.0026
Mesh effects	-0.0008
Temperature increase to 200°F	+0.0110
Pertubation of infinite array by different spacing between modular assemblies	+0.0032
Axial neutron leakage	-0.0020
Total pertubation on basic cell reactivity calculation	~ +0.0194

4.0 SEISMIC ANALYSIS

The new spent fuel racks for Oyster Creek are designed for a maximum seismic event of .31 g horizontal acceleration applied simultaneously with normal (1.0 g) gravity plus or minus .31 g vertical acceleration. The earthquake loads on the rack and base structures are calculated on the basis of the largest, fully-loaded spent fuel rack configuration. The direction of the horizontal seismic component is assumed to be in that worst-case direction which results in the maximum loads at any fuel rack corner joint. In addition, each fuel box is designed to accommodate two fuel assemblies for a total design weight of 1,374 pounds.

The safe shutdown earthquake of the Oyster Creek plant was based on a 0.22 g maximum ground acceleration in accordance with Section 3.1.1 of the FDSAR. The response of the containment building at various locations, including the first three modes of building vibration, rocking, and higher modes, was given in a revised reactor building seismic analysis (by John A. Blume & Associates, San Francisco, CA, June 18, 1965) for various elevations. Since the elevation of the spent fuel pool floor is 80' 6" but the building responses are tabulated for 75' 3" and 95' 3", a linear interpolation of reactor building seismic acceleration values is used to determine the proper SSE acceleration of 0.312 g at the spent fuel pool floor. It has been determined that the mass of proposed additional spent fuel has an insignificant effect on the reactor building dynamic response, natural frequencies or mode shapes. The increase in fuel pool floor loading due to weight of additional fuel elements does not exceed the original design floor load considered for contribution of stored fuel elements.

The effect of water sloshing in the fuel pool was also analyzed, and it was determined that the sloshing effect is not significant compared to the seismic loads on either the fuel racks or pool structures.

The spent fuel racks are classified Seismic Category I in accordance with USNRC Regulatory Guide 1.29. They are designed for and will withstand the seismic loadings previously described. The honeycomb-like stainless steel structure of the rack not only provides a smooth, all-welded stainless steel box to protect the fuel assembly and preclude seismic damage, but also serves as a neutron absorber and will maintain the fuel in a non-critical (nuclear) array so long as the stainless steel box wall surrounds the fuel assembly.

Each stainless steel box is securely fastened to its neighbors. The resulting honeycomb-like structure is quite stiff; in fact, the rack is much stiffer than the support base I-beams by an order of magnitude. The I-beams serve as the load path to transfer seismic loads from the rack to the pool floor and walls, and also provide a redundant rack support structure. The stainless steel box configuration and thickness was selected on the basis of nuclear requirements as well as convenience in handling, shipping, stability, and resistance to low frequency vibrations.

The loaded spent fuel rack (which includes water inertial effects and assumes that each cell contains two spent fuel assemblies and the base structure is capable of withstanding accident loads, including the Oyster Creek OBE and DBE seismic requirements. When the rack structure was subjected to the DBE (.31 g) load at the spent fuel pool level, the stresses of all applicable structural components did not exceed the following AISC limitations:

a. Tension or compression over a gross sect[illegible] $\sigma_T \leq 0.60\ \sigma_y$ $\sigma_c \leq 0.60\ \sigma_y$

where σ_y is the 0.2% yield strength of the stainless steel.

b. Shear over gross section $\tau_s \leq 0.40\ \sigma_y$

c. Bending stresses - tensile and compressive $\sigma_b \leq 0.66\ \sigma_y$

d. Buckling stresses - compression only $\sigma_c \leq 0.60\ \sigma_{CR}$

where σ_{CR} is the lowest load critical buckling stress.

e. Tension or compression on solid round or square bars; also for bending stress or solid rectangular bars about weaker axis

$\sigma_{ST} \leq 0.75\ \sigma_y$

$\sigma_{SC} \leq 0.75\ \sigma_y$

$\sigma_{RB} \leq 0.75\ \sigma_y$

Recognizing that yield stress σ_y and elastic modulus, E_y, are functions of temperature, both properties were extracted from tables in Section III of the 1974 ASME Boiler and Pressure Vessel Code. The temperature at which these properties have been selected is assumed to be 200°F. Since the water in the spent fuel pool is not expected to reach 200°F for any appreciable time, the values used for σ_y and E_y were conservative.

Weld materials are generally considered to be identical to the base material since full-strength welds will be made in accordance with the AWS recommended sizes. In any case, all crucial structural welds were designed to and limited by the following stress values:

f. Groove weld - tensile stress $\sigma_y \leq 0.74\ \sigma_{all}$

g. Groove weld - shear stress $\sigma_{WS} \leq 0.60\ \sigma_{all}$

h. Fillet weld - shear stress $\sigma_{FS} \leq 0.49\ \sigma_{all}$

where all three stress limits and allowable limit value σ_{all} were extracted from tables in Section VIII of the 1974 ASME Code. The shear stress limit for fillet welds was generally less than the shear stress over gross section limitation and, therefore, was conservative.

The results to be presented are based on the following:

a. The racks are made from Type-304 stainless steel which has a minimum yield strength (0.2%) of 30,000 psi and a minimum tensile strength of 75,000 psi at room temperature. The values used in the stress and vibration analyses assume the pool temperature to be ≤ 200°F which results in a yield strength of 25,000 psi and an elastic modulus of 27.7 million psi.

b. The entrained water from the horizontal and vertical motion occupies all the rack space at water box locations and 0.6 of the rack space at the spent fuel location.

c. No benefit is taken for the horizontal friction forces between the bottom of the rack leveling pads and the pool floor.

d. No benefit is taken for the dampening effect of the water.

The actual stress values calculated and the results of the seismic vibration analysis are:

a. Lowest fundamental (first mode cantilever vibration) horizontal natural frequency of fully-loaded 4 x 2 configuration of fuel racks = 38.8 Hz.

b. The seismic acceleration taken from the "Earthquake Acceleration Response Spectrum", Figure V-3-1 of the FDSAR which corresponds to 38.8 Hz = 0.312 g.

c. Lowest fundamental (first mode simply-supported beam vibration) vertical natural frequency of fully-loaded 4 x 2 configuration of fuel racks = 78.6 Hz.

d. The seismic acceleration taken from the "Earthquake Acceleration Response Spectrum", Figure V-3-1 of the FDSAR which corresponds to 78.6 Hz = 0.312 g.

e. Horizontal (4 x 2) Seismic Weight = 239,843 lbs. (minimum).

f. Vertical (4 x 2) Seismic Weight = 239,843 lbs. (minimum).

g. Submerged (4 x 2) Dead Weight = 189,488 lbs. (minimum).

h. Compressive Stress in Box Walls:

σ_c = 2376 psi for 0.312 g horizontally, 1.312 g vertically (SSE)

σ_c = 1189 psi for 0.156 horizontally, 1.156 vertically (OBE)

σ_c = 282 psi for 0.0 horizontally, 1.0 g vertically (Normal)

$0.6\ \sigma_y$ = 15,000 psi

$0.6\ \sigma_{cr}$ = 19,518 psi

i. Tensile Stress in Existing Swing Bolts (worst case for 4 x 2 rack):

σ_T = 7,485 psi maximum calculated load

$0.75\ \sigma_y$ = 18,750 psi design limit

Since the rack design meets the required seismic criteria, a seismic event will not alter the k_{eff}. The design permits only one fuel assembly to be inserted into a storage cell and the structure maintains minimum center-to-center spacings (the criticality calculations accounted for tolerances and minimum spacings).

Note also that this planned modification does not change the analysis of the effect of potential missiles on the spent fuel pool as presented in Amendment 32 of the FDSAR (Question 12). This concludes that no perforation of the spent fuel pool floor will occur from any postulated missiles striking it.

The above analyses and controls give a high level of confidence that no undue risk will result from these new fuel racks during their installation or during a seismic event.

5.0 SPENT FUEL POOL COOLING

5.1 The spent fuel pool cooling system and shutdown cooling system are described in the Oyster Creek FDSAR, Volume I, Section X, and are shown diagramatically in Volume II, Figures X-2-2 and X-3-2. For convenience, a brief description of the spent fuel pool and shutdown cooling systems is given below:

5.1.2 Spent Fuel Pool Cooling System

The fuel pool cooling system removes decay heat from spent fuel which has been removed from the reactor and is in the spent fuel storage pool. The cooling system maintains the storage water temperature low enough to prevent steaming and to maintain good visibility.

The fuel pool cooling system also 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.

The fuel pool cooling system is designed to normally operate at 125°F; however, it may, at times, reach 140°F which is the temperature limit of the demineralizer resin.

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 pools into the surge tanks. Return flow goes into the spent fuel pool and also into the reactor cavity through two

return diffusers mounted on the cavity wall in the east and west regions, about four feet above the reactor flange.

5.1.3 Shutdown Cooling System

The purpose of the reactor shutdown cooling system is to remove decay heat from the reactor during shutdown operations. This operation is accomplished by circulating the reactor water through the shutdown system heat exchangers where decay heat is removed. Three pump and heat exchanger combinations are provided with a sufficient capacity to remove the decay heat being generated in the core twenty-four hours after shutdown. An interlock prevents opening the isolation valves between the recirculation system and this system when the reactor pressure is above 150 psig.

5.1.4 Reactor Building Closed Cooling Water System

Both the spent fuel pool cooling system and the shutdown cooling system are cooled by the Reactor Building Closed Cooling Water System (RBCCWS). The RBCCW system is comprised of two heat exchangers and two pumps. Inhibited demineralized water is circulated through the system and is cooled by the Service Water System.

The RBCCWS is designed not only to cool the SDC and FPC systems but many reactor plant components. The RBCCW system is rated at 1.16×10^8 BTU/hr. at 133°F inlet with a service water temperature of 85°F.

5.2 Determination of Heat Load on Spent Fuel Pool

The decay heat generation rate for the spent fuel pool was calculated using decay heat generation curves developed by M. J. Bell. The Bell analysis was based on data from References 17, 18, and 19.

Bell's data pertain to 3.3 w/o PWR fuel irradiated for 1100 days. Fission products from U-235, U-238, and Pu-239 are included as is the contribution of the actinides. Figure 8 is the decay heat curve developed by Bell showing a comparison to the ANS 18.6 curve. Figure 9-A, B, and C, is the decay heat curve presented in the NRC Standard Review Plan.

As can be seen, the Bell curve is more conservative than the decay heat curves presented in both ANS 18.6 and NRC Standard Review Plan, "Technical Position APCSB.9-2". The Bell curve correlates well with 120% of ANS standard, and therefore, accounts for uncertainty predictions established in ANS 18.6.

In performing calculations to determine the maximum heat load in the spent fuel pool, the following power history, representing the worst case core unload condition, was assumed.

Power History

At the time the core was unloaded, the spent fuel pool contained nine, one-quarter core batches (140 elements per batch), one batch having been unloaded each year for nine consecutive years. It is also assumed that each one-quarter core batch was irradiated for four, forty-six week periods, interrupted by a six-week outage once a year, prior to unloading. Therefore, the decay heat from spent fuel results from nine, four-year quarter cores, the first batch unloaded decaying for nine years, the second batch unloaded for eight years, etc. through the last batch unloaded which has been decaying for one year. It was then assumed that the core was unloaded after reactor operation at full power for a period of forty-six weeks (normal refueling period) resulting in four, quarter core batches irradiated for approximately one, two, three, and four years, respectively.

Decay heat calculations were done starting at a time ten days after reactor shutdown since that is considered to be the minimum time necessary to unload the core to the spent fuel pool and replace the gate between the spent fuel pool and reactor cavity. The resultant decay heat is shown in tabular form in Table V-1 and a heat generation curve in Figure 10. The maximum heat generation rate at ten days after shutdown is 18.76×10^6 BTU/hr.

A significant result of the decay heat rate calculations is that at ten days after shutdown the oldest seven, quarter core batches in the fuel pool contribute only 5% of the total heat load to the spent fuel pool. The majority of the heat (95%) results from the full core unload plus the last two quarter cores to be unloaded (i.e., the present spent fuel pool capacity of 840 elements).

To further illustrate the minimal effect the proposed spent fuel pool expansion would have on heat load in the spent fuel pool, Figure 10 shows a heat generation curve for a full core unload with the present Oyster Creek spent fuel pool capacity. Also plotted on Figure 10 are heat generation curves for normal refuelings for existing spent fuel pool capacity and the proposed expanded capacity. These curves show that if the spent fuel pool was filled to the increased capacity under a normal refueling schedule, the resulting total heat generation would be less than 10% of that which would result from refueling to the present capacity.

It can be concluded from the above discussion that expansion of the spent fuel pool capacity from 840 to approximately 1800 storage spaces

does not significantly increase the heat load to the spent fuel pool cooling system, the RBCCW system, nor the service water system.

5.3 Spent Fuel Pool Cooling Capacity

The Oyster Creek Spent Fuel Pool Cooling system is presently rated at 5.5×10^6 BTU/hr. with a fuel pool temperature of 125°F and RBCCW system temperature at 90°F. This rating is based on 10% heat exchanger tubes plugged and no fouling. Clean heat exchangers with no tubes plugged results in a spent fuel pool cooling system rating of 7.08×10^6 BTU/hr. at 125°F fuel pool temperature and 90°F RBCCW temperature. There are no tubes plugged in either fuel pool heat exchanger at this time.

Each spent fuel pool heat exchanger has identical tube-side and shell-side flow rates. Consequently, the mean temperature difference for the exchanger is simply the arithmetic average of the inlet temperature difference ($T_{\text{tube inlet}} - T_{\text{shell inlet}}$), and the outlet temperature difference ($T_{\text{tube outlet}} - T_{\text{shell outlet}}$). It is, therefore, possible to express the combined cooling capacity of the heat exchangers directly as a function of the inlet temperature difference and reference design conditions:

$$Q_t = \frac{2\,Q_{ref}}{(T_{TI} - T_{SI})_{ref}} \times (T_{TI} - T_{SI})$$

Where Q_t = the total cooling capacity for two fuel pool heat exchangers operating at T_{TI} and T_{SI}.

Q_{ref} = the design cooling capacity as specified on one HX data sheet.

T_{TI} = the fuel pool water temperature (tube-side inlet temperature).

T_{SI} = the cooling water supply temperature (shell-side inlet temperature).

$(T_{TI} - T_{SI})_{ref}$ = the difference between the tube-side inlet and shell-side inlet temperatures at the design cooling capacity.

It has been calculated that the fuel pool cooling system capacity increases by approximately 2.02×10^6 BTU/hr. for each 10°F decrease in sea water temperature or increase of fuel pool temperature. It

has also been calculated that through the normal range of operation there is approximately a 5°F temperature differential between service water and RBCCW. Using the above calculated value, it can be seen that with an expected low service water temperature of 35°F the fuel pool cooling system capacity would increase to approximately 20.22×10^6 BTU/hr. without exceeding the spent fuel pool temperature of 140°F.

Figure 11 is a family of curves showing the effect of service water temperatures on fuel pool temperature versus time after shutdown. It should be noted that these curves are conservative since they do not take into account the cooling effect of evaporation or heat up of structural members and various other metal items in the pool.

5.4 Thermohydraulics

The coolant return flow to the spent fuel pool is discharged via two spargers entering from the corners of the south wall and running north 8 feet along the east and west walls. This will tend to fill the spaces between the walls and fuel racks where the spargers are first, then the like space along the south wall, and probably even the center north-south space between the two groups of fuel racks on either side of the pool sump trench.

Water will flow from the wall spaces to between the base supports beneath the rack boxes. From here, the water will flow toward the center of the pool, with varying amounts branching to feed the various fuel assemblies. As shown in Figure 2, the fuel assemblies will be 11 inches above the pool floor. This is considered more than adequate flow area. Several large holes are also cut into the base I-beam web to allow circulation in all directions. Each fuel assembly's flow will depend upon its heat dissipation rate and the total pressure loss experienced by the base flow reaching its inlet (lower nozzle) location. This depends upon other fuel assembly heat rates and flows. The flows from all the fuel assemblies mix above the fuel racks and move toward the north wall weirs. The forced circulation via the fuel pool cooling system and the natural circulation "chimney" effects in the box-type fuel cells provides greater flow through the hotter elements. This is primarily due to the decrease in water density around the hotter fuel elements which results in a lower ΔP through the racks.

It has been calculated that sufficient flow is provided to the hottest fuel element to prevent its cladding temperature from exceeding the local fuel pool water saturation temperature of approximately 240°F.

5.5 Procedural Control to Prevent Excess Pool Temperature

In the unlikely event that it becomes necessary to unload a full reactor core, procedural control will be established using the temperature relationship discussed above to maintain the pool temperatate at ≤ 140°F.

The time after shutdown at which the core can be unloaded and the fuel pool gates closed, isolating the FPC system from the SDC system is dependent on actual discharge history and power history of the reactor or decay heat generation rate and service water cooling temperature. Since these variables are known or can be conservatively estimated at the time the decision is made to unload a full core into the spent fuel pool, the delay time for unloading the core, which will not allow the fuel pool temperature to exceed 140°F, can be calculated with confidence. This is presently the procedure to be used by Jersey Central Power & Light Company should core unload be necessary.

During a normal refueling outage, the unloading of a one-quarter core need not be delayed, even to the maximum capacity of the proposed expansion, since the heat generation rate at ten days after shutdown with twelve quarter cores in the pool is approximately 6.3×10^6 BTU/hr.

5.6 Expanded Cooling System

The fuel pool cooling system and shutdown cooling system were originally designed with capped connections for a cross connect from the FPC system to the "A" heat exchanger of the SDC system. This cross connect results in approximately double the present fuel pool cooling system capacity.

Jersey Central Power & Light Company will proceed with the installation of the cross connect in order to minimize delays in unloading the core should it ever be necessary. It is expected that the cross connect will be installed by mid-1977 and upon its completion, the curves of Figure 11 will be modified accordingly. The piping will be designed and erected in accordance with the requirements of Regulatory Guides 1.26 and 1.29.

5.7 Thermal Inertia

The thermal inertia of the fuel pool has been calculated as 2.48×10^6 BTU/°F. If the spent fuel pool cooling system becomes inoperative at the time a full core is unloaded ten days after shutdown, it would take 9.5 hours before the fuel pool would boil assuming an initial pool temperature of 140°F. This is considered sufficient time to either make repairs or add make-up

cooling water from the make-up system or an outside source such as the fire hoses. Question 12 of Amendment 32 to the FDSAR discusses the pool makeup capability and notes that normal makeup water to the pool is provided from the nominal 5.25×10^5 gallon condensate storage tank at a rate of about 250 gpm by a single condensate transfer pump. The makeup capability from this system is increased to about 420 gpm if both condensate transfer pumps are used. Additional makeup at a rate of 150 gpm can be provided from the nominal 3×10^4 gallon demineralized water storage tank by the demineralized water transfer pumps through hose connections to service connection boxes in the pool area. The two skimmer surge tanks which handle pool level surges contain a total of about 3500 gallons normally, or up to 7000 gallons if full, which can be pumped into the pool at a rate of 1000 gpm by the fuel pool cooling pumps. The 2000 gpm diesel driven fire pumps can be used to provide makeup water to the condensate storage tank through the permanent connection to the tank.

5.8 Conclusions

On the basis of the discussion presented above, the following conclusions can be drawn in regard to spent fuel pool cooling.

5.8.1 The service water system and the Reactor Building Closed Cooling Water System are both capable of handling the increased heat generation of approximately 1×10^6 BTU/hr. contributed by an additional 96? spent fuel elements in the spent fuel pool under steady-state conditions.

5.8.2 For a normal refueling outage in which a quarter core is unloaded, there are no additional restrictions which must be imposed with an expanded capacity.

5.8.3 The Spent Fuel Pool Cooling System capability depends upon service water temperature and time after shutdown. Procedural control will be used to insure that a full core will be unloaded to the spent fuel pool only after it has been determined that the spent fuel pool temperature will not exceed 140°F.

5.8.4 The cooling capability of the fuel pool cooling system will be augmented (approximately doubled) by cross connecting to the "A" heat exchanger of the shutdown cooling system within the next eighteen months.

TABLE V-1
OYSTER CREEK DECAY HEAT RATES, 10^6 BTU/HR
(ESTIMATED BELL DATA)

1/4 CORE NO.	DAYS AFTER SHUTDOWN**										
	10	12	16	20	25	30	45	60	75	100	150
1	3.928	3.684	3.258	2.934	2.630	2.376	1.842	1.530	1.325	1.073	.934
2	4.222	3.976	3.544	3.214	2.904	2.644	2.093	1.764	1.543	1.277	.981
3	4.346	4.099	3.667	3.336	3.024	2.762	2.208	1.876	1.650	1.378	1.075
4	4.402	4.156	3.724	3.394	3.082	2.821	2.264	1.931	1.706	1.433	1.132
5	.621	.617	.610	.601	.592	.582	.556	.530	.504	.467	.427
6	.325										
7	.219										
8	.171										
9	.136										
10	.117										
11	.102										
12	.092										
13	.082										
TOTAL	18.76	17.77	16.05	14.72	13.47	12.43	10.21	8.87	7.97	6.87	5.79

NOTES: *1/4-Cores "1, 2, 3, & 4" are 1, 2, 3, & 4 years old, respectively. "5" is a 4-year 1/4-core decayed 1 year, "6" a 4-year 1/4-core 2 years, etc.

**After a normal 46-week 100% power operating period.

6.0 INSTALLATION

6.1 The new spent fuel racks will be installed in the Oyster Creek spent fuel pool on an "as needed" basis to insure that there will always be sufficient space to unload a full core. It is anticipated that an initial set of 24 racks (12-28 element plus 12-32 element) will be installed prior to the 1977 refueling outage in order to provide full core unload capability.

6.2 The spent fuel racks are installed in two phases; first the base supports (Figure 4) are installed and leveled and then the racks (Figure 1) are positioned one at a time on the base support and secured to them. Each of the racks are tied to adjacent racks after installation to form a minimum grouping of eight racks. This has previously been discussed in Sections 2.0 and 4.0 and shown on Figure 3.

The rack base assemblies are designed to be bolted down using the existing swing bolts in the bottom of the pool. By extending the length of these base assemblies, but maintaining the same hold-down pattern, it is possible to bridge into areas where there are no tie-down bolts, therefore, providing the capability to fully utilize all available space in the spent fuel pool.

6.3 The installation of new racks must be accomplished with water in the spent fuel pool. Although the racks incorporate design features, which allow installation, assembly, and tie down using remote tools and methods, it is anticipated that divers would be used to facilitate cleanup, removal of old racks, and new rack installation. To minimize the radiological affect on the divers, spent fuel will be moved to remote areas of the pool prior to their entry. A detailed radiological survey will be performed, and the divers will be equipped with several dose rate and integrating dose devices. The divers will also be equipped with special suits to preclude contact with spent fuel pool water.

6.4 It has been calculated that if one rack assembly, weighing approximately 4500 pounds, were dropped from the pool sil (40 feet above pool floor), the spent fuel pool floor integrity would not be affected.

Although dropping a single compacted spent fuel rack does not jeopardize the pool floor integrity, several precautions will be taken during installation to preclude the possibility of dropping a rack.

As noted, only one rack will be installed at a time, each rack having a total weight of 4500 pounds.

The racks will be lifted over the pool with an overhead bridge crane with a 100-ton rated hook. The crane will be load tested prior to installing racks with a weight at least double the maximum expected load.

All existing spent fuel racks in the area where a new rack is to be installed will be unloaded and all spent fuel will be placed in a remote area of the pool. The new racks will then be lifted over the empty racks and translated to the position where they are to be installed. This will minimize the effects of dropping a load since the empty racks would collapse to absorb much of the energy. Secondly, when the new rack is moved out over the pool floor, its maximum height above the floor would only be about 15 feet.

6.5 Figure 13 shows the location of spent fuel racks in the Oyster Creek Pool presently. Figure 14 shows the proposed layout of compacted spent fuel racks at the completion of the installation.

Since these analyses show that the proposed installation procedure does not jeopardize the integrity of the spent fuel pool floor, the installation poses no unreviewed safety question.

7.0 RADIOLOGICAL CONSEQUENCES

As mentioned in the Introduction, there were 326 assemblies in the spent fuel pool as of January 1976. This number is derived from the 270 assemblies that were in the pool just prior to the December 1975 refueling plus the 56 offloaded assemblies from this refueling. The 270 assembly inventory was the result of fuel shipping prior to the refueling outage. The contractual agreement for this shipping has been fulfilled, and there are no other shipping arrangements. Therefore, assuming a yearly refueling, it is estimated that the spent fuel inventory will be 540 assemblies by 1980. No shipping arrangements are planned prior to 1980.

Table VII-1 shows recent fuel pool water isotopic analysis data. Note that these data show Co^{60}, Cs^{134}, and Cs^{137} concentrations after shutdown but prior to fuel offload into the spent fuel pool for the Spring-1975 refueling (i.e., April 4, 1975) when there were 368 assemblies in the pool. The July 3, 1975 data show the concentrations after the Spring-1975 refueling (some assemblies had been shipped by July 3). The September 16, 1975 data show the concentrations about 2-1/2 months after the refueling when approximately 350 assemblies were in the spent fuel pool. There were slight increases in the concentrations after the refueling offload. The largest increases were in the Cesium concentrations just after the offload. Note that they had decreased significantly by the September analysis. Recent improvements in fuel design have markedly decreased the number of leaking fuel pins. Because of these data and fuel design improvements, radionuclide concentrations in the expanded capacity fuel pool should present no operational difficulties.

Figure 12 shows the results of a radiation survey of the spent fuel pool taken in mid-November 1975 (3 feet above the surface). Note that the radiation levels are relatively low with the higher measurements the result of stored LPRM's, channel clips, etc. The radiation levels from the expanded capacity pool will be closely monitored and are not expected to pose any operational difficulties. Area radiation monitors provide continual surveillance.

As shown in Section 4.0, above, the new rack design, which is structurally stronger than the present racks, is designed to meet the necessary seismic requirements and to protect the spent fuel against dropped assemblies. The cask drop analysis for possible fuel damage is unchanged, and the pool increase, therefore, poses no greater risk from this consideration. Because of these, and the fact that the pool temperature will not be allowed to exceed 140°F, no additional radioactive releases beyond those previously analyzed are expected because of this modification.

The above considerations and results give a high level of confidence that no deleterious radiological consequences will result from the proposed modification.

Note that even though the assembly density and total number of assemblies in the fuel pool are increased, increasing the probability of a fuel pin rupture due to a drop accident, the expected number of ruptured fuel pins from such an accident is certainly bounded by the accident analysis assumptions of Section B-XIII of Amendment 65 of the Oyster Creek FDSAR. Here the possible rupture of 445 fuel pins was considered from a fuel assembly drop on the core, which has a greater assembly density than the proposed installation. It was shown that the whole body and thyroid doses are well below the 10CFR100 limits.

These assumptions are extremely conservative for possible spent fuel pool accidents and do show that the planned modification poses no increased risk to the public. Note also that the effect of Oyster Creek's new 8x8 fuel design on the assembly drop accident was addressed in Section V.6 of Amendment 76 to the Oyster Creek FDSAR, dated January 31, 1976. This shows that the previous analyses are conservative. In fact, the gaseous fission product inventory is less in the 8x8 design than in the older 7x7 design.

Although these analyses show that the increased assembly density in the spent fuel pool has increased the probability of rupturing a fuel pin due to a drop accident, it has not exceeded the accident analysis assumptions of Section B-XIII of Amendment 65 of the Oyster Creek FDSAR. Therefore, this analysis of radiological consequences poses no unreviewed safety questions.

TABLE VII-1

PLANT STATUS	SAMPLE DATE	COUNT DATE & TIME	Co^{60} µCi/ml	Cs^{134} µCi/ml	Cs^{137} µCi/ml
1. Shutdown (84 hrs) Just Prior to Sipping	4-01-75	4-01-75 @ 2232	1.3×10^{-4}	1.9×10^{-4}	2.9×10^{-4}
2. During Operation	7-03-75 @ 0820	7-03-75 @ 1045	2.4×10^{-4}	8.3×10^{-4}	1.5×10^{-3}
3. During Operation	9-16-75 @ 1430	9-16-75 @ 1520	2.7×10^{-4}	2.9×10^{-4}	5.9×10^{-4}

8.0 COST-BENEFIT ANALYSIS

Typically, the Oyster Creek Nuclear Generating Station is refueled once a year. Each refueling replaced about one-quarter of the core (about 140 assemblies), and each new assembly contains about 175,000 grams of uranium.

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. Consideration was given to possible storage in the spent fuel pool of the Metropolitan Edison Company's Three Mile Island Unit 1 (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.

Storage at commercial storage facilities has also been evaluated. It has been determined that the average cost, including transportation, for such storage is approximately $3,620/year/assembly. This means that only about 415 assemblies could be shipped before this alternative becomes less economical (considering rack modification only). It is obvious that these alternatives would be economical only for a short term (i.e., two to three years) solution. The state of spent fuel reprocessing dictates that prudent management should pursue a longer term solution. This then determines that the onsite storage capacity expansion is the most desirable solution. Of course, the extreme alternative would be shutdown of the Oyster Creek Station when the storage capability was no longer available. This consideration is obviously unacceptable since the current energy replacement value for Oyster Creek is approximately $360,000 a day (assuming 620 MWe).

The pursuit of the Oyster Creek spent fuel pool capacity modification clearly seems the most reasonable from a cost/benefit view.

9.0 CONCLUSION

9.1 The safe operation of the Oyster Creek Nuclear Generating Station is necessary for the production of energy at reasonable rates for the well being of the people of New Jersey. The continued safe operation requires that the spent nuclear fuel be stored for future processing. This capability will soon be lost if provisions are not made for expanded storage capacity or shipping. Several storage and shipping alternatives were considered as discussed above, and these considerations determined that the expansion of the Oyster Creek Station's spent fuel pool capacity is clearly the most attractive from a cost-benefit view. Pursuit of this determination has led to a storage rack modification design that meets all applicable requirements even when including extremely conservative assumptions. It has been shown that the proposed modification does not pose any undue risk to the health and safety of the public nor does it constitute any significant environmental impact.

9.2 The proposed modification and its installation involves an unreviewed safety question as defined in 10 CFR 50.59 because Section 5.3.B of the Oyster Creek Technical Specification must be changed to specify a k_{eff} of stored spent fuel in the spent fuel pool of .95 in lieu of the present limit of .90 (see below).

The proposed modification and its installation reduces the margin of safety as defined in the Basis for Technical Specification 5.3.B. However, the new margin is consistent with the appropriate standards and design criteria as delineated previously. No other margin of safety, as defined in the Basis for any Technical Specification has been reduced.

The proposed modification and its installation does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

The proposed modification and its installation does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report because:

a. The proposed installation procedure does not jeopardize the integrity of the spent fuel pool floor as discussed in Section 6.0.

b. The increased assembly density in the fuel pool would be bounded by the accident analysis assumptions of Section B-XIII of Amendment 65 of the Oyster Creek FDSAR, as discussed in Section 7.0.

10.0 FIGURES

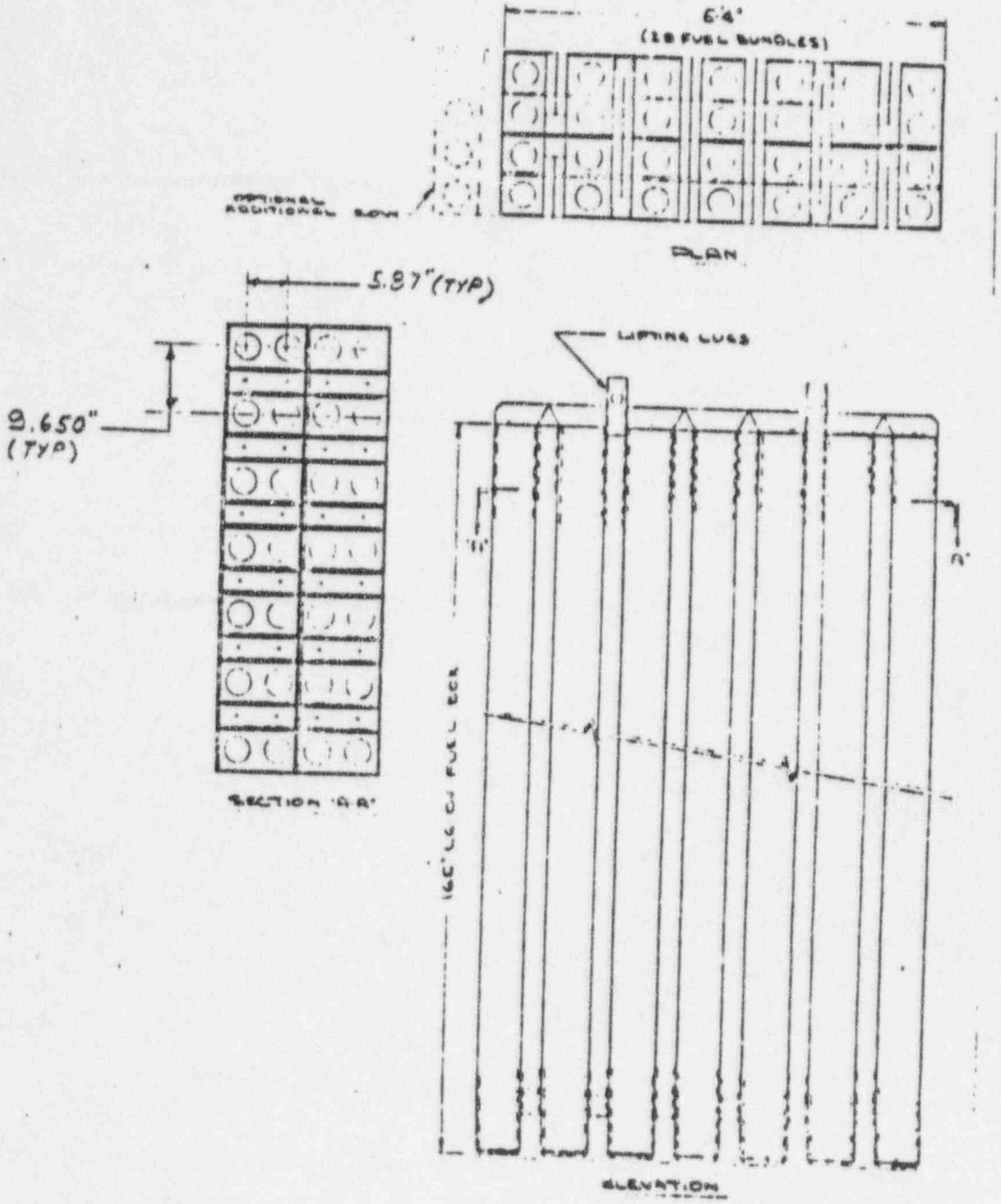


FIGURE 1- RACK ASSEMBLY

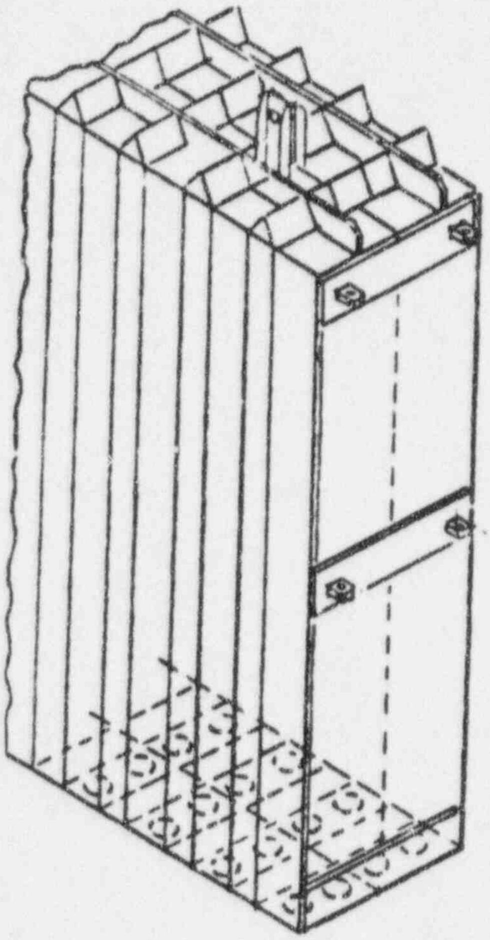


FIGURE 2 FUEL ASSEMBLY LEAD IN CONFIGURATION

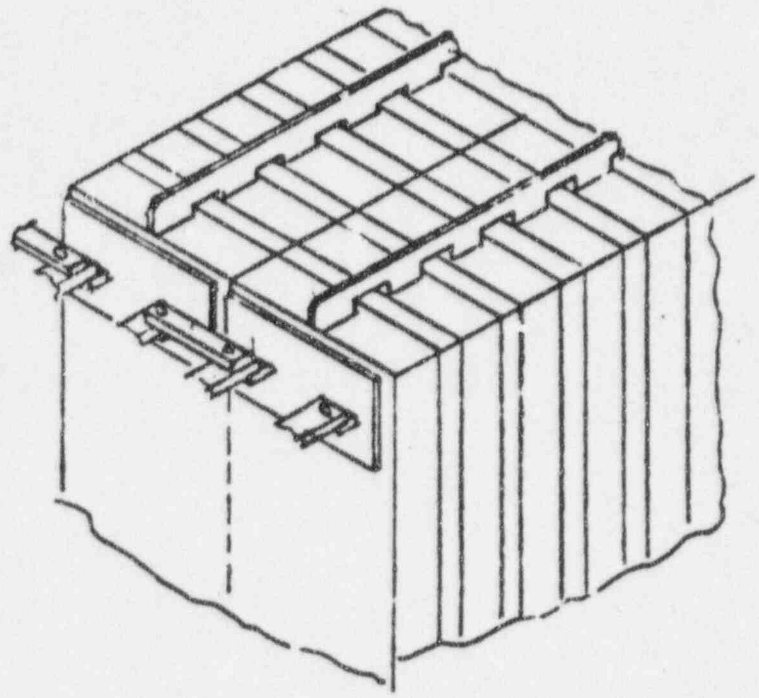


FIGURE 3 MECHANICAL CONNECTIONS BETWEEN RACK ASSEMBLIES

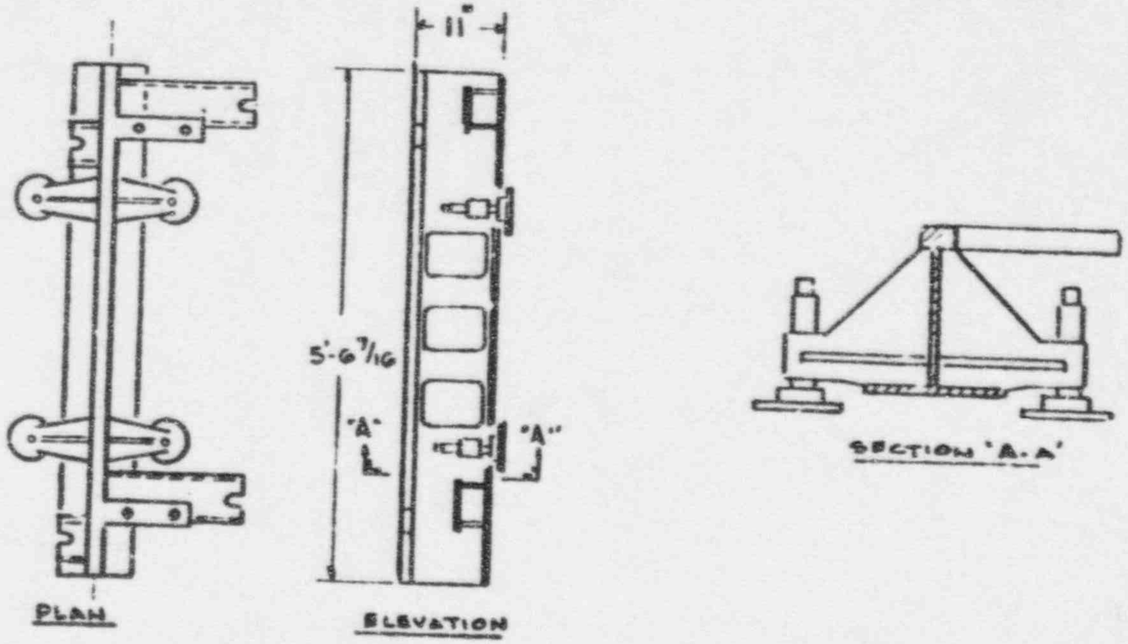


FIGURE 4 BASE ASSEMBLY

FIGURE 5

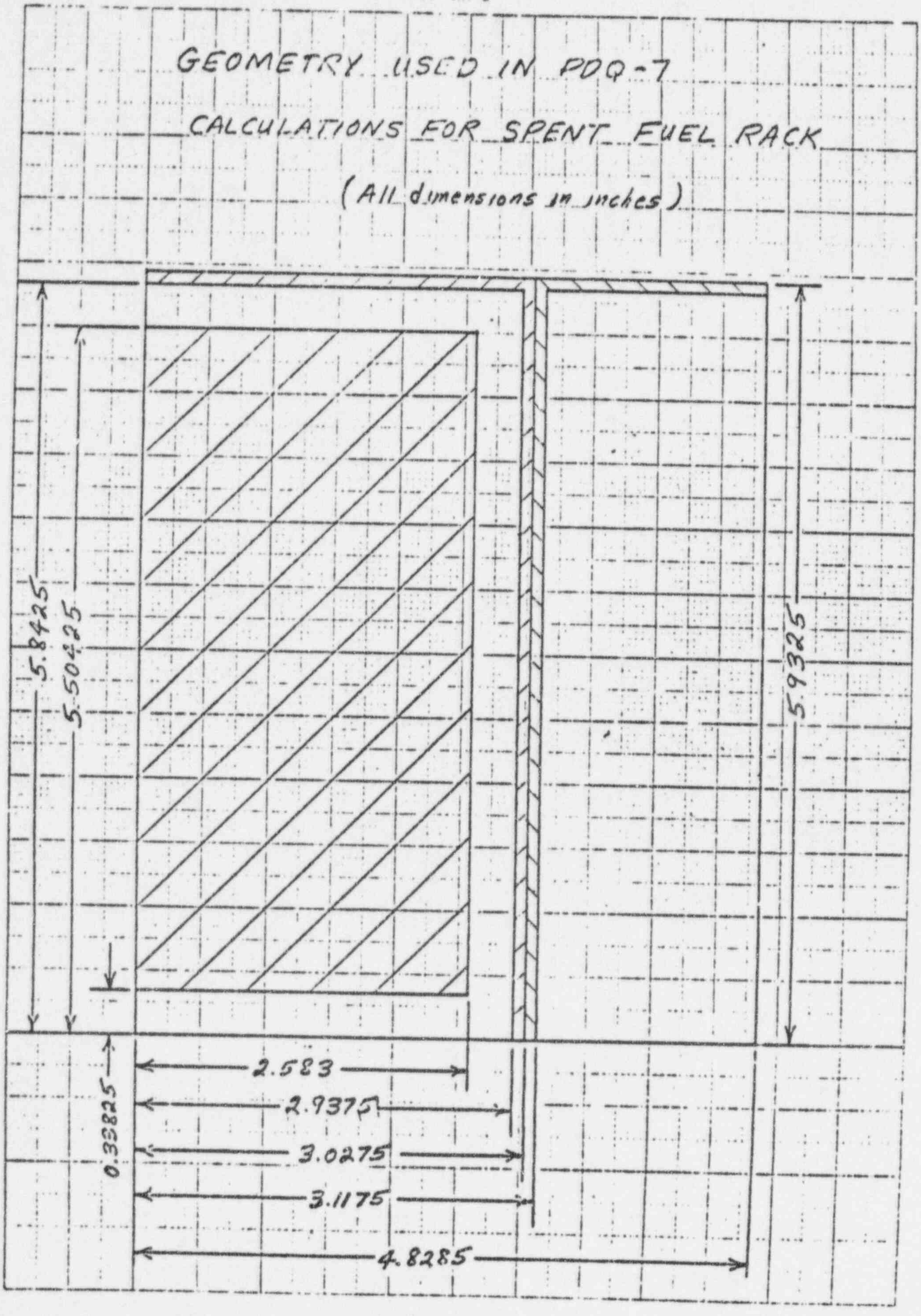
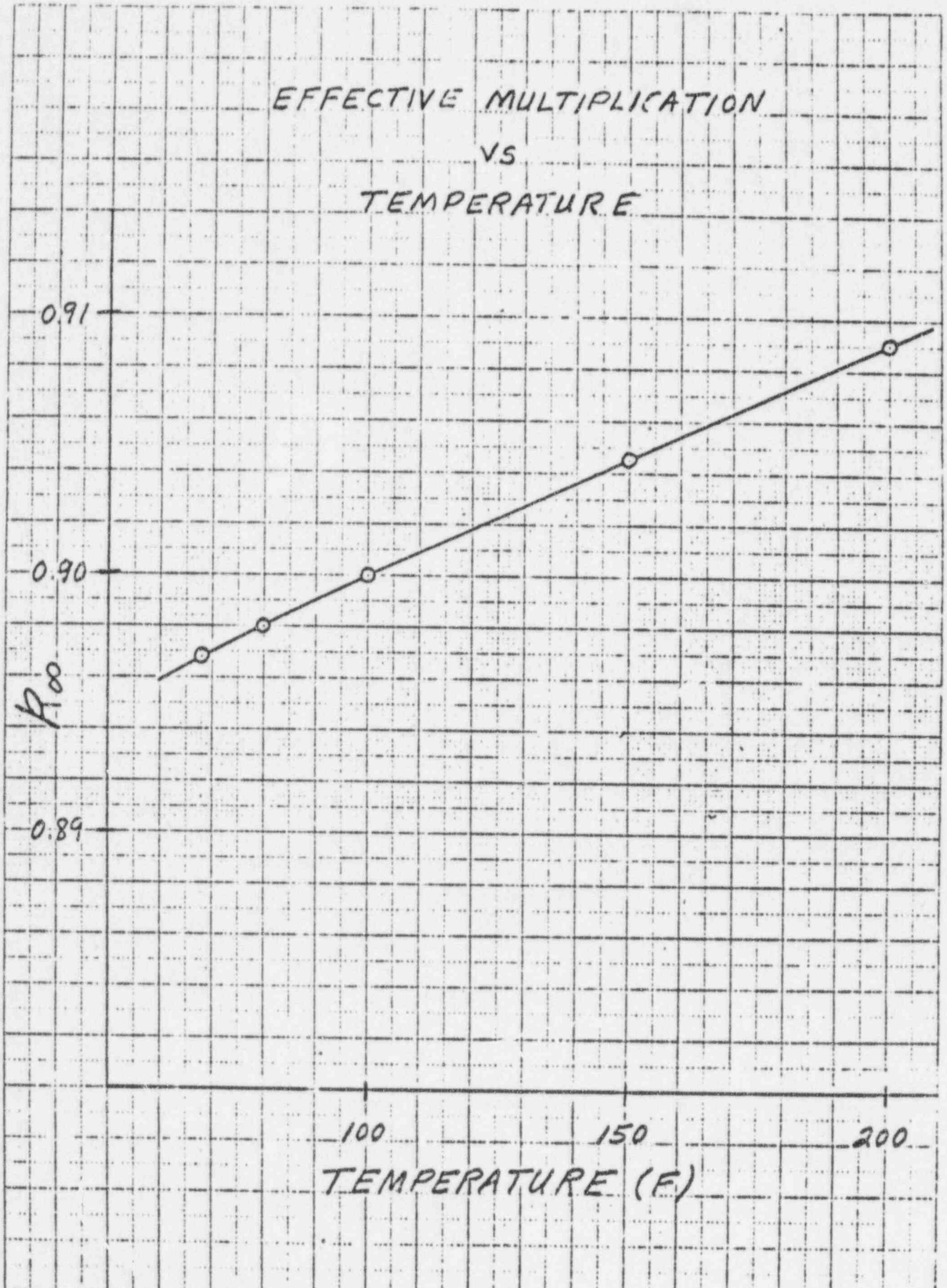

GEOMETRY USED IN PDQ-7
CALCULATIONS FOR SPENT FUEL RACK
(All dimensions in inches)
5.8425
5.50425
5.9325
0.33825
2.583
2.9375
3.0275
3.1175
4.8285

FIGURE 6

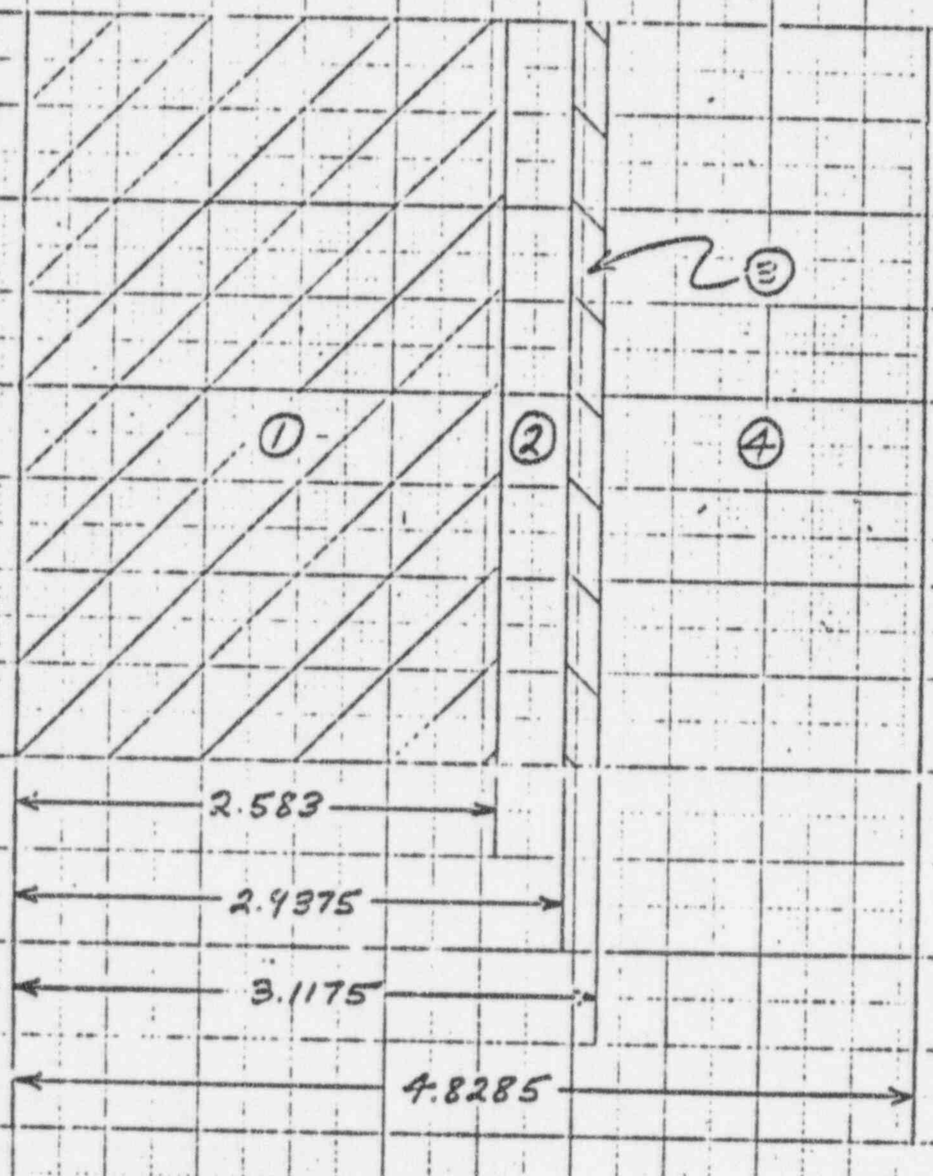


EFFECTIVE MULTIPLICATION
VS
TEMPERATURE
0.91
0.90
0.89
k
100
150
200
TEMPERATURE (F.)

FIGURE 7

ONE-DIMENSIONAL, FLUX WEIGHTED BASIC CELL

REGION NO.	MATERIAL
1	HOMOGENIZED FUEL, WATER, & SS
2	HOMOGENIZED WATER & SS
3	EXPLICIT SS
4	HOMOGENIZED WATER & SS



All Dimensions in inches

K+E LOGARITHMIC 5 X 5 CYCLES
KEUFFEL & ESSER CO.

FIGURE 3

DECAY HEAT (watts/watt)

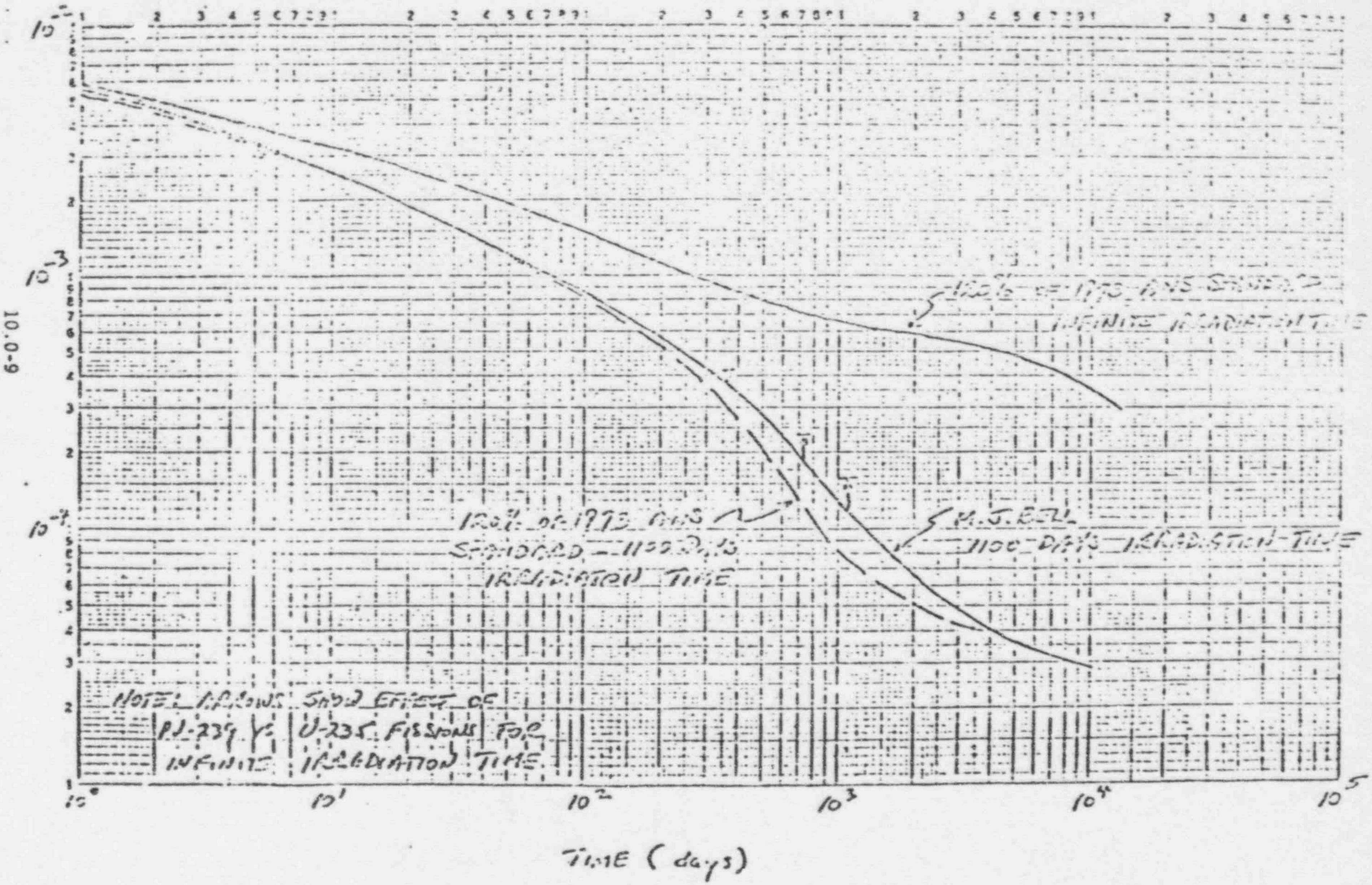


FIGURE 9a

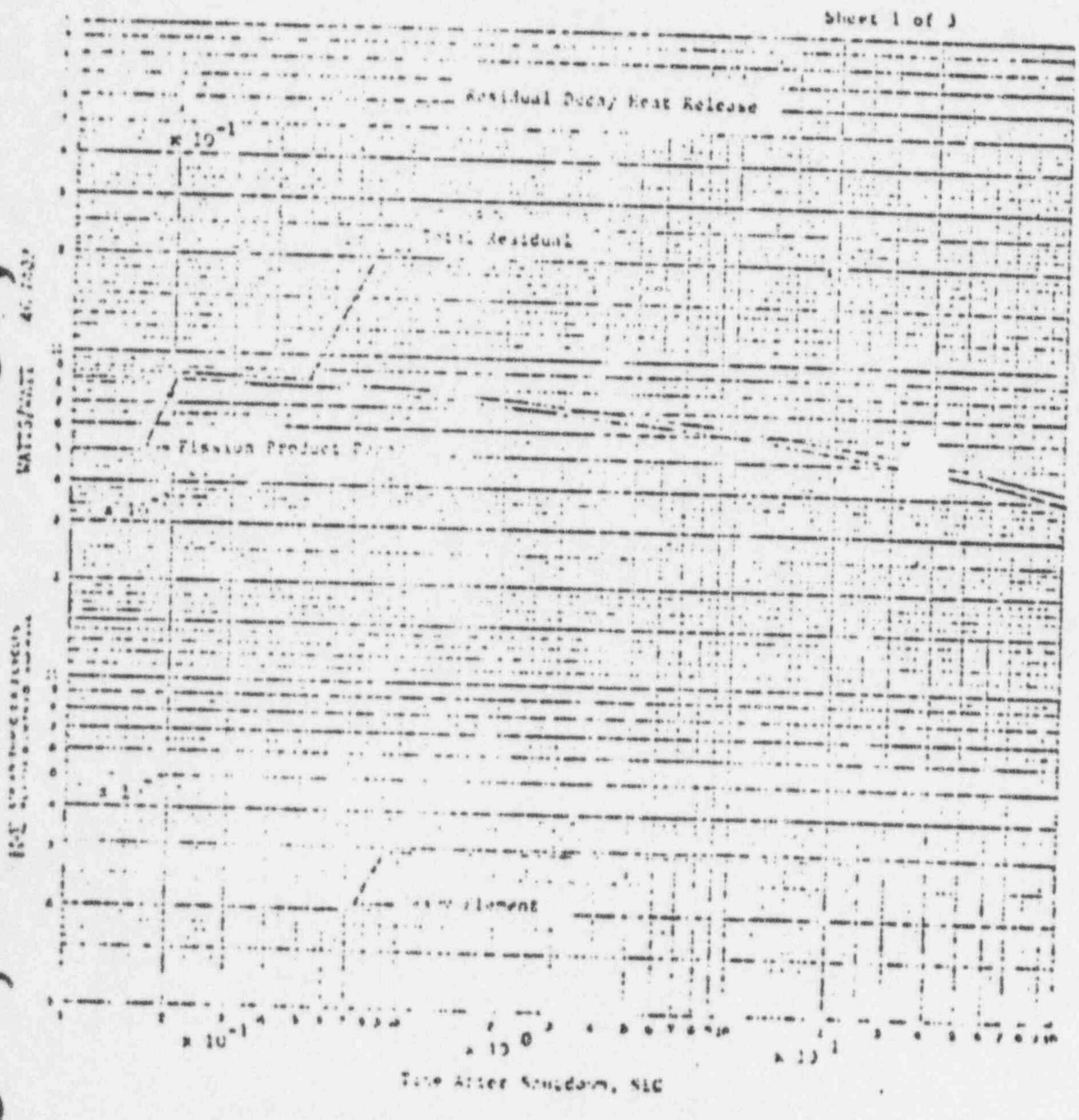

Sheet 1 of 3
Residual Decay Heat Release
Total Residual
Fission Product
Time After Shutdown, SEC

FIGURE 9b

Sheet 2 of 3

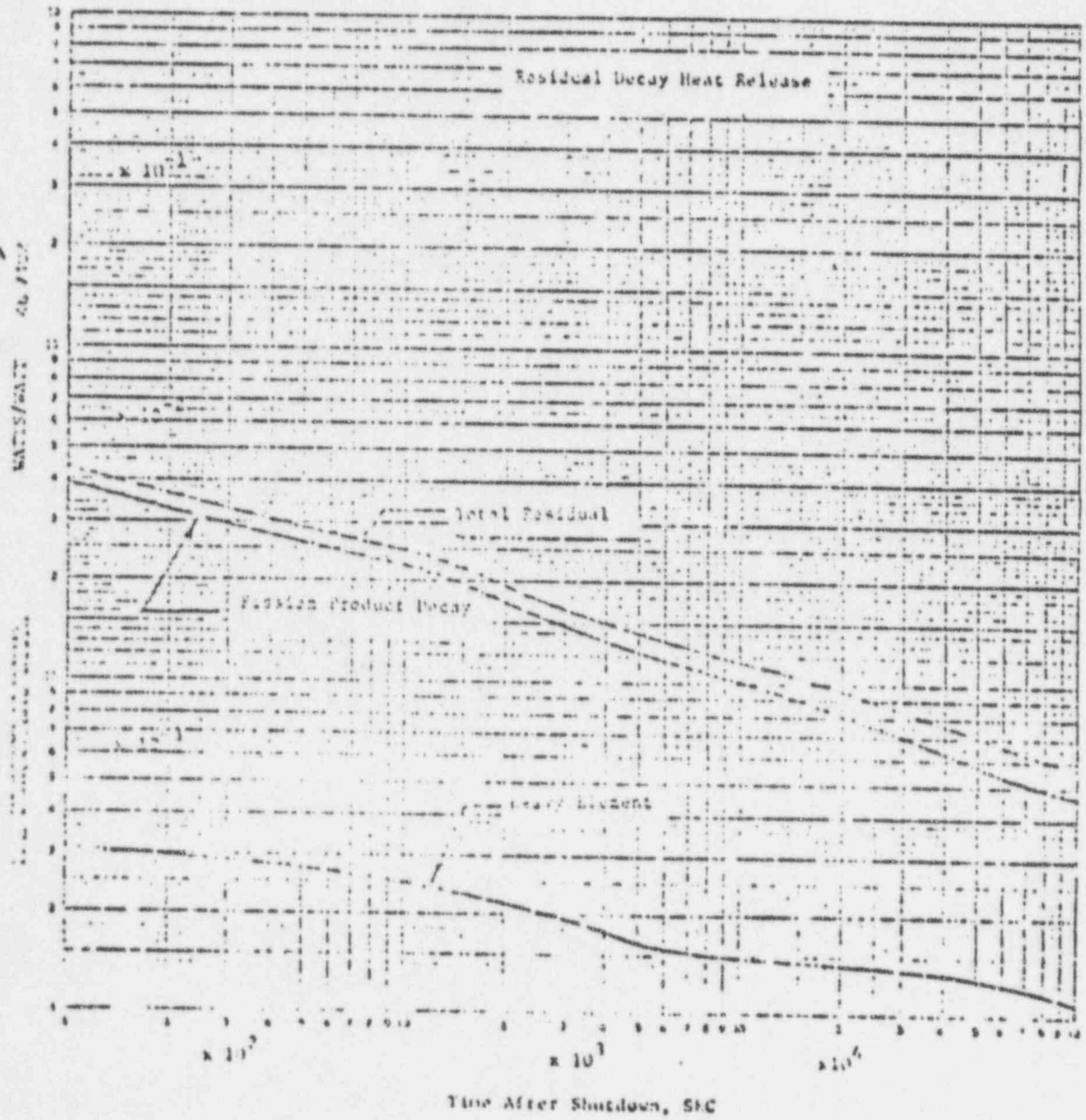
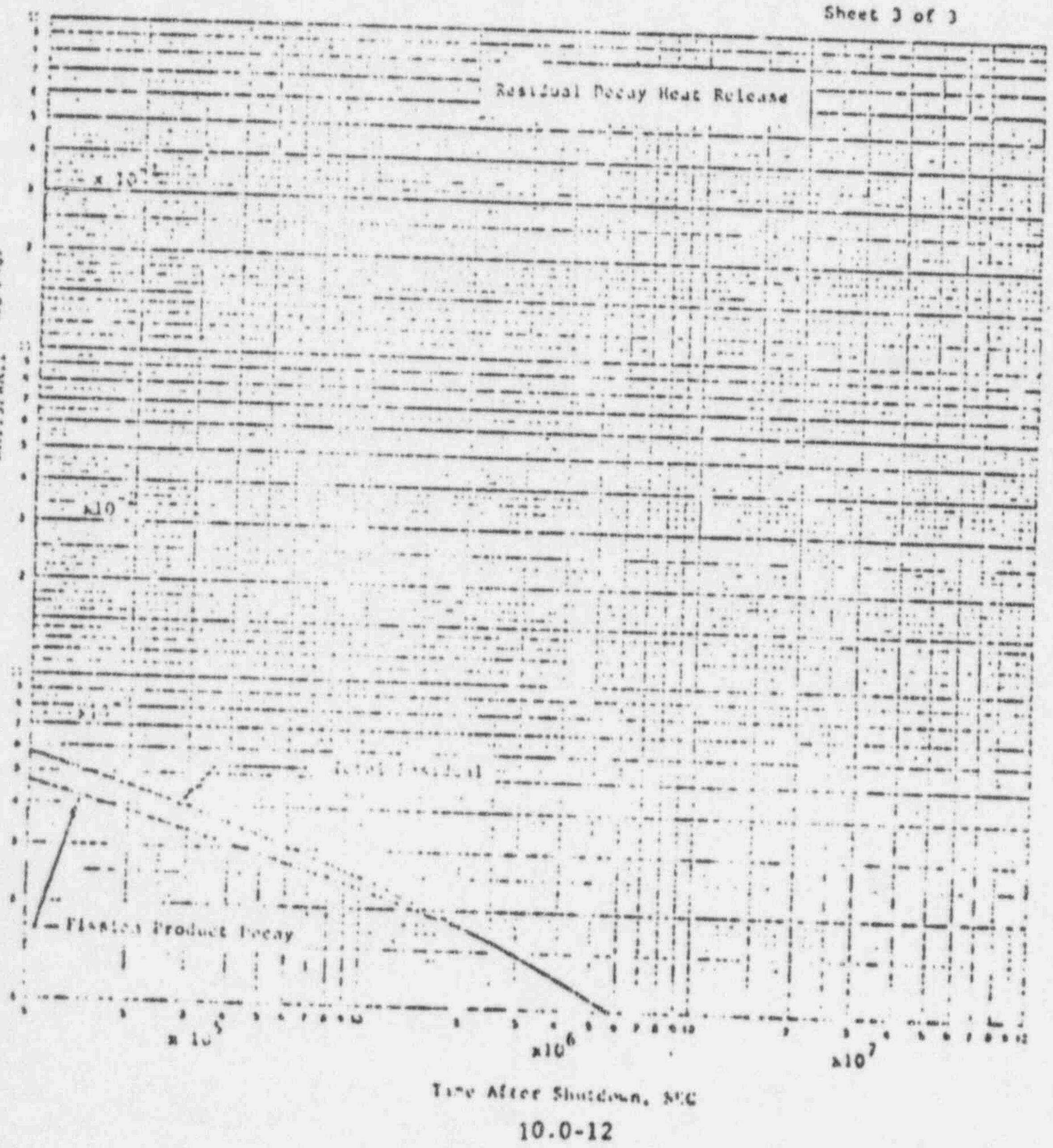
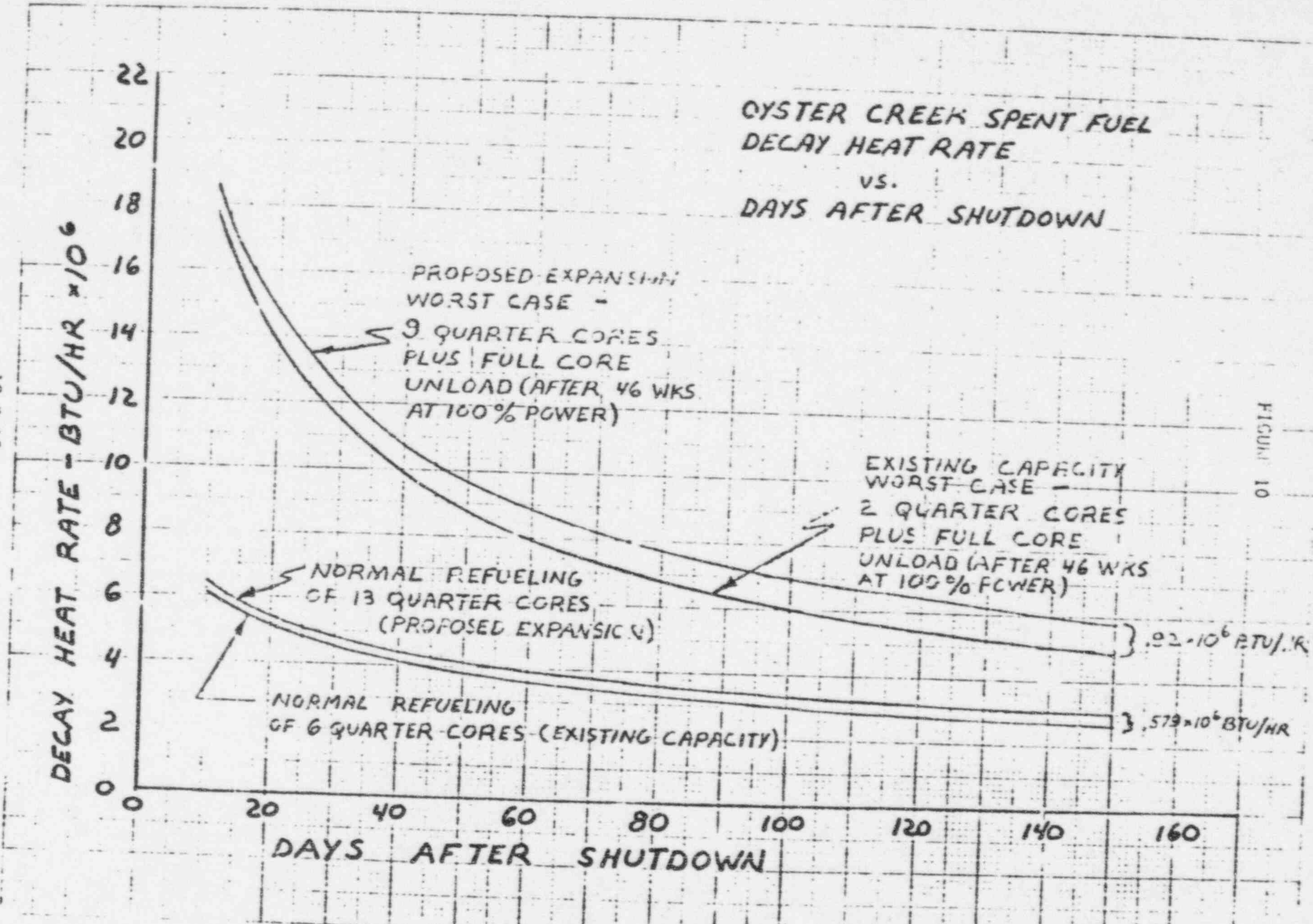


FIGURE 9c

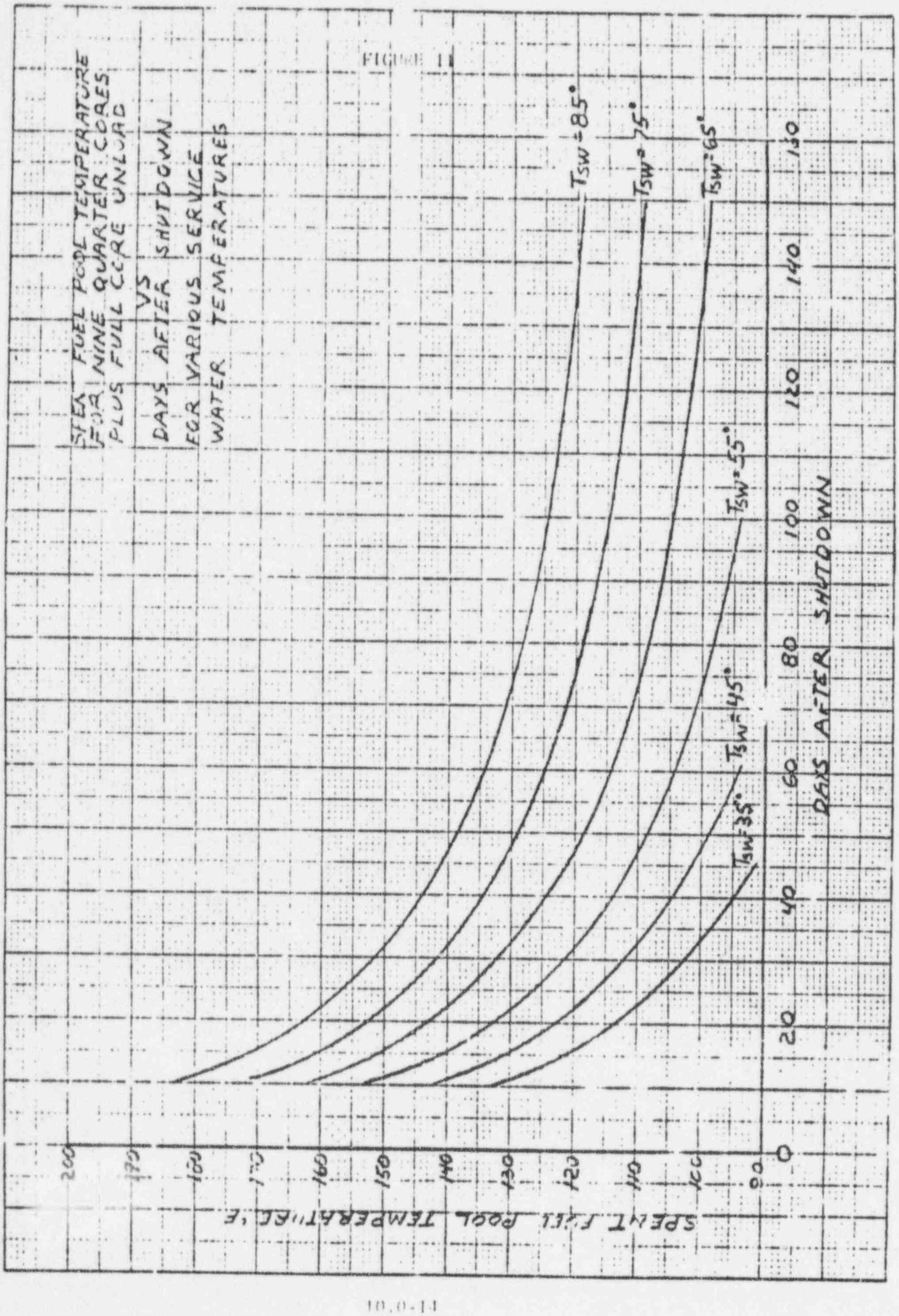

Sheet 3 of 3
Residual Decay Heat Release
Total Residual
Fission Product Decay
x 10^5
x10^6
x10^7
Time After Shutdown, SEC



10.0-13

FIGURE 10

FIGURE 11


SPENT FUEL POOL TEMPERATURE
FOR NINE QUARTER CORES
PLUS FULL CORE UNLOAD
VS
DAYS AFTER SHUTDOWN
FOR VARIOUS SERVICE
WATER TEMPERATURES
TSW=85°
TSW=75°
TSW=65°
TSW=55°
TSW=45°
TSW=35°
DAYS AFTER SHUTDOWN
0
20
40
60
80
100
120
140
160
SPENT FUEL POOL TEMPERATURE °F
200
190
180
170
160
150
140
130
120
110
100
90

SPENT FUEL POOL
RADIATION SURVEY (11-12-75) MREM

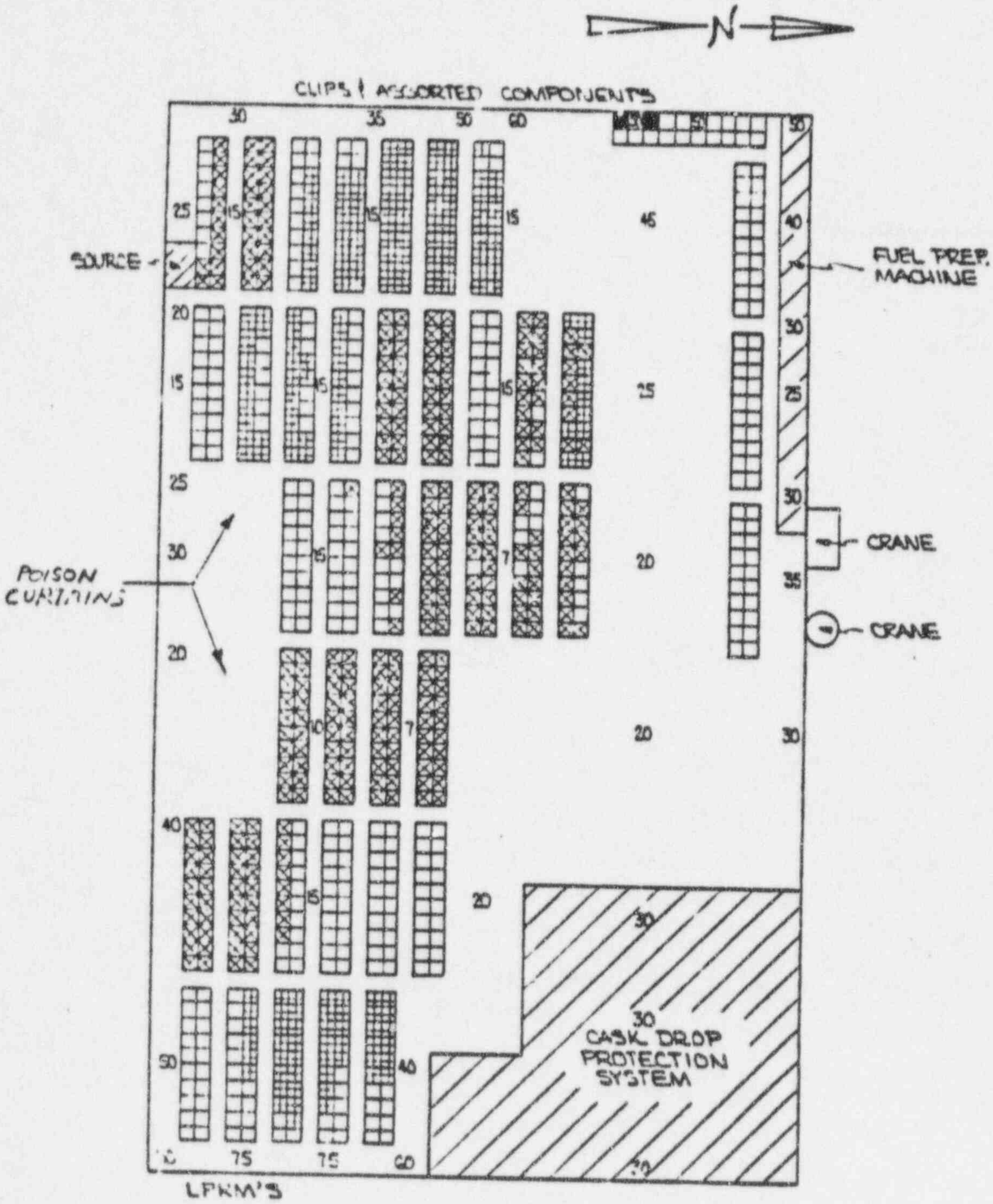
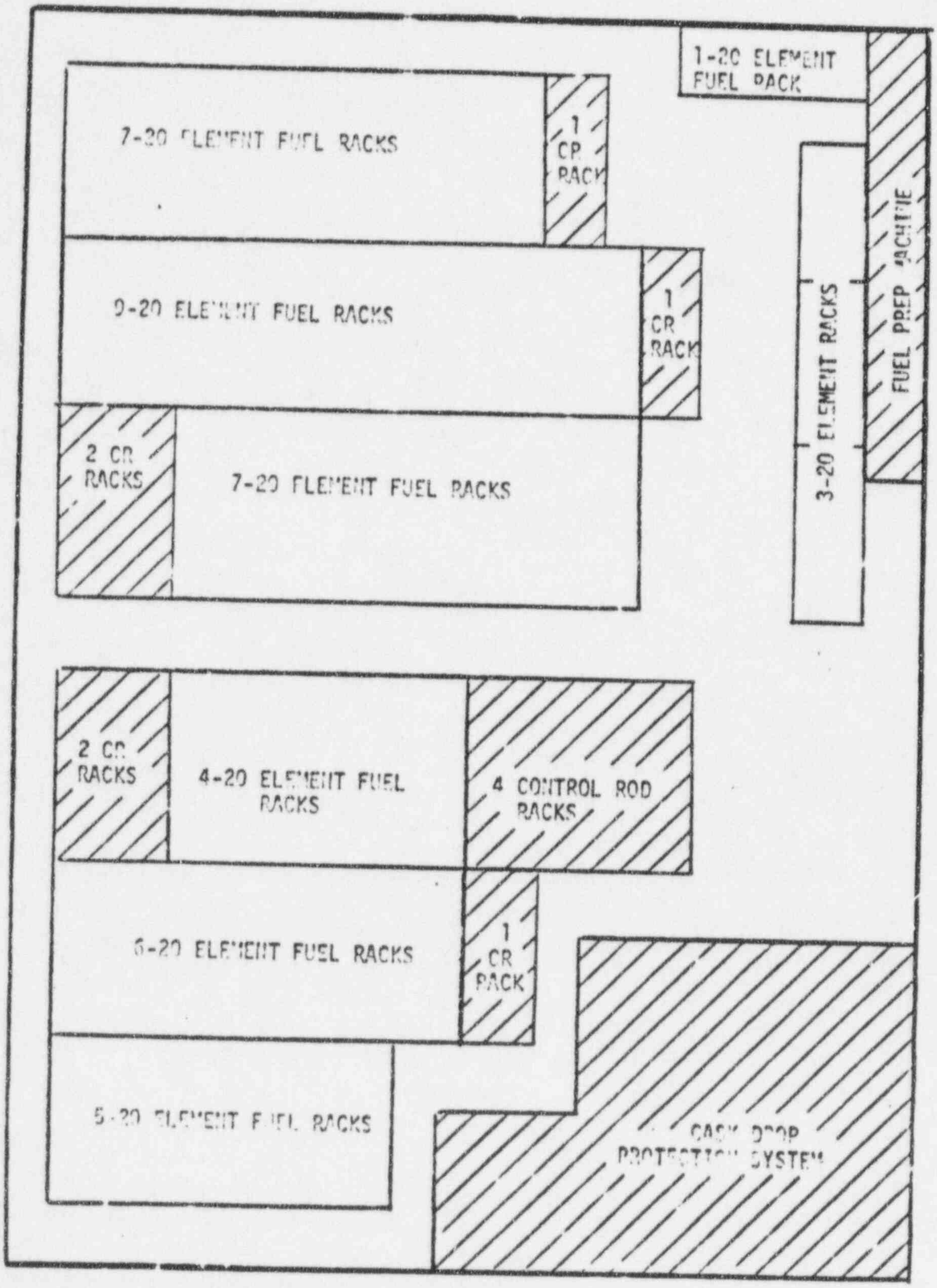


FIGURE 12

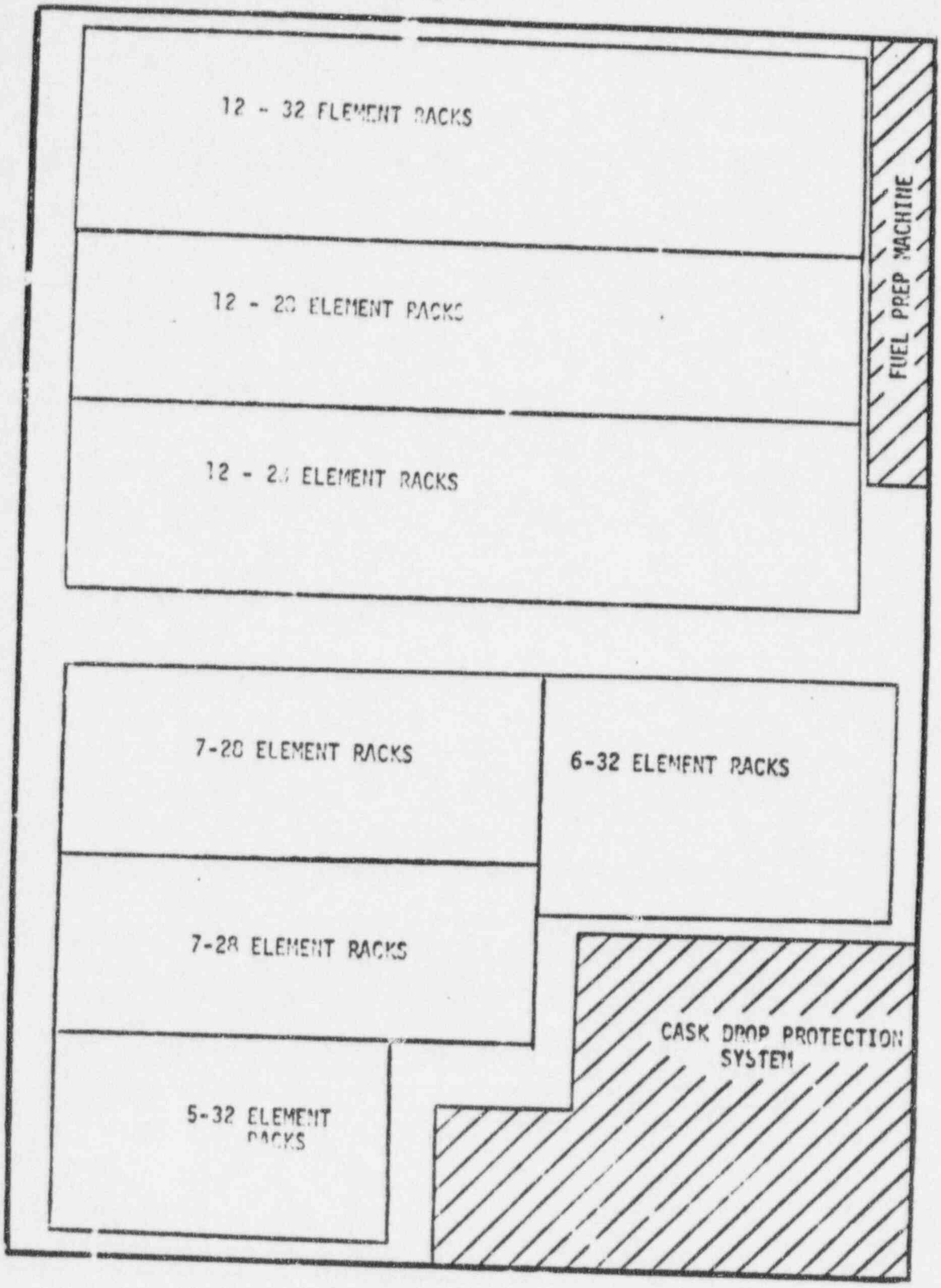
▦ FUEL PIN CONTAINER
\+ CHANNEL
X SPENT FUEL ASSEMBLY

FIGURE 13



EXISTING LAYOUT-OC
SPENT FUEL POOL

FIGURE 14



PROPOSED LAYOUT-OC
SPENT FUEL POOL

11.0 REFERENCES

1. R. F. Barry, "LEOPARD -- A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094", WCAP-3269, September, 1963.

2. W. R. Cadwell, "PDQ-7 Reference Manual", WAPD-TM-678, January, 1967.

3. H. Bohl, E. Gelbard, and G. Ryan, "MUFT-4 -- Fast Neutron Spectrum Code for the IBM-704", WAPD-TM-72, July, 1957.

4. H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code", WAPD-TM-39, January, 1957.

5. L. E. Strawbridge and R. F. Barry, "Criticality Calculations for Uniform Water-Moderated Lattices", Nuclear Science and Engineering, 23, 58, 1965.

6. "Large Closed-Cycle Water Reactor Research and Development Program Progress Report for the Period January 1 - March 31, 1965", WCAP-3269-12.

7. "List of Equipment and Apparatus at WREC", Westinghouse Reactor Evaluation Center, February, 1967.

8. W. L. Orr, H. I. Sternberg, P. Deramaix, R. H. Chastain, L. Bindler, and A. J. Impink, "Saxton Plutonium Program, Nuclear Design of the Saxton Partial Plutonium Core", WCAP-3385-51, December, 1965. (Also EURAEC-1490).

9. R. D. Leamer, W. L. Orr, R. L. Stover, E. G. Taylor, J. P. Tobin, and A. Bukmir, "PuO_2-UO_2 Fueled Critical Experiments", WCAP-3726-1, July, 1967.

10. A. F. Henry, "A Theoretical Method for Determining the Worth of Control Rods", WAPD-218, August, 1969.

11. P. W. Davison et al., "Yankee Critical Experiments - Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water", YAEC-94, Westinghouse Atomic Power Division (1959).

12. V. E. Grob and P. W. Davison et al., "Multi-Region Reactor Lattice Studies - Results of Critical Experiments in Loose Lattices of UO_2 Rods in H_2O", WCAP-1412, Westinghouse Atomic Power Division (1960).

13. W. J. Eich and W. P. Kovacik, "Reactivity and Neutron Flux Studies in Multiregion Loaded Cores", WCAP-1433, Westinghouse Atomic Power Division (1961).

REFERENCES (Continued)

14. W. J. Eich, Personal Communication (1963).

15. T. C. Engelder et al., "Measurement and Analysis of Uniform Lattices of Slightly Enriched UO_2 Moderated by D_2O-H_2O Mixtures", BAW-1273, The Babcock & Wilcox Company (1963).

16. A. L. MacKinney and R. M. Ball, "Reactivity Measurements on Unperturbed, Slightly Enriched Uranium Dioxide Lattices", BAW-1199, The Babcock & Wilcox Company (1960).

17. Tabular Data for the Standard Decay Heat Curve, Table I, ANS 18.6 Standard on Decay Heat, 1973 version.

18. Calculated Radiation Properties of Spent Plutonium Fuels, M. J. Bell, ORNL, Nuclear Technology, Volume 18, April, 1973.

19. Generic Environmental Statement Mixed Oxide Fuel (WASH-1327), Table IV-H-3 and IV-H-5, USAEC, August, 1974.

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

MASTER

June 24, 1976

Exhibit 5

Docket No. 50-219

Docket - 50219 - - 930

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

Gentlemen:

By letter dated March 18, 1976, you presented Amendment No. 78 to the FDSAR and proposed changes to the Technical Specifications for the Oyster Creek Nuclear Generating Station. These proposed amendments would incorporate the modifications and specifications necessary for the planned increase in spent fuel storage capacity of the spent fuel storage pool at the Oyster Creek Station.

In order to continue our evaluation of your proposed changes, response to the enclosed request for additional information must be provided. The response or a schedule for providing the response should be forwarded within 30 days. Please contact us if you have any questions on our request.

Sincerely,

George Lear

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

Enclosure:
Request for Additional
Information

cc: See next page

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cc:

G. F. Trowbridge, Esquire
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Barr Building
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Washington, D. C. 20006

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
1712 N Street, N. W.
Washington, D. C. 20036

Honorable Joseph W. Ferraro, Jr.
Deputy Attorney General
State of New Jersey
101 Commerce Street - Room 208
Newark, New Jersey 07102

Ocean County Library
15 Hooper Avenue
Toms River, New Jersey 08753

The Honorable W. M. Mason
Mayor, Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Mark L. First
Deputy Attorney General
36 West State Street
Trenton, New Jersey 08625

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

REQUEST FOR ADDITIONAL INFORMATION
OYSTER CREEK NUCLEAR GENERATING STATION
SPENT FUEL POOL - INCREASED STORAGE CAPACITY
DOCKET NO. 50-219

1. Provide sketches of the fuel pool storage racks which define the primary structural aspects and elements relied upon for the structure to perform its safety function. Include typical details of all interfaces with the pool boundaries and connections between rack assemblies. Also provide sketches of the fuel storage pool showing its principal dimensions and structural features and its relationship with surrounding structures and supports. Include sketches of the existing "swingbolts" and illustrate how racks will be placed in areas where there are no swingbolts.

2. Provide sketches of the mathematical model of the fuel pool, fuel storage rack and fuel assembly system which will be used in the analysis and discuss the effects of submergence in water. Illustrate on the sketches the mechanism of shear and load transfer to the fuel pool walls and foundation slab. Discuss the possible impact of the fuel assemblies with the rack and describe how the effect of water sloshing in the fuel pool was analyzed.

3. Provide specific information on the loads, load combinations and acceptance criteria which will be utilized in the design of the racks. Identify the magnitude of all loads considered in the design.

4. Describe the design and analysis procedures for the fuel storage rack, including the expected behavior under load and the mechanism of load transfer to the foundation. Discuss the resistance of the racks to sliding and overturning. Computer programs should be referenced to permit identification with available published programs.

5. Discuss the extent to which the fuel pool has been analyzed to verify its ability to withstand the increase in overall loading. Identify the loads and load combinations investigated and the acceptance criteria for concluding that the original structure is adequate.

6. Identify all the materials to be used in the fabrication and construction of the fuel pool racks. Describe the extent to which you intend to comply with ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants".

7. Indicate whether appropriate damping values and combination of modes and spatial excitation will be in accordance with Regulatory Guides 1.61 and 1.92 respectively for the analysis of the fuel pool and the fuel storage rack seismic system. It is noted that Table II-1 references R.G. 1.92; however the discussion in the first paragraph of Section 4.0 is not consistent with the guide.

8. The discussion in the second paragraph of Section 4.0 implies that the building response at the elevation of the spent fuel pool floor was used to prescribe the seismic loading on the racks. However, the discussion on the bottom of Page 4.0.3 indicates that the floor response spectrum at this elevation was used. Provide a detailed discussion to clarify the design approach.

9. It is indicated in the report that the fundamental frequency of the racks is greater than 33 Hz in both vertical and horizontal modes of vibration. Provide mode frequencies, mode shapes and participation factors for the first few modes to substantiate your position. The relevant dynamic model should also be presented. The staff position is that at the high frequency end, the ground response spectrum may not have any amplification over the maximum ground acceleration but contribution from significant modes should be considered for overall response.

10. Compare the most severe temperature distribution with that used in the original structural design of the fuel pool structure. Justify any increase in the maximum design basis transient temperature.

11. Discuss the extent to which the behavior of each storage rack assembly was analyzed when fastened to other assemblies within the fuel pool. Discuss the effect on the rack design when only some of the assemblies are loaded with fuel.

12. Discuss the ability of the rack to withstand the loadings imposed by a postulated dropped fuel assembly. Describe the loads and acceptance criteria and the design and analysis procedures utilized in the design. Include a discussion of the maximum drop height considered in the design, the masses involved, the kinetic energy at the point of impact and the amount of ductility utilized to dissipate the kinetic energy of the impact.

13. Describe what controls will be exercised to ensure that the specified clearances between the fuel racks and the pool walls will be maintained during and after installation.

14. The three inches stated for the width of the flux trap region on page 2.0-1 and the 5'4" given for the length of the rack for 28 fuel bundles as given on page 10.0-2 are not consistent with the 9.650" storage lattice pitch given on the same page. Also, the 4.8285" half lattice pitch given on page 10.0-6 (Fig. 5) is not exactly consistent with the 9.650" lattice pitch given on page 10.0-2 (Fig. 1). Please provide a consistent set ot numbers and drawings.

15. In data submitted for reload cycle #5 the overall dimension of the type VB fuel bundle is given as 5.217". But the overall fuel region size given in Figure 5 of FDSAR Amendment 78 is 5.166". What was the basis for choosing this 5.166" dimension for your calculations and what is the overall dimension of the maximum sized fuel bundle or assembly that will be put into this storage rack?

16. It appears that most of the critical experiments which you calculated consist of a uniform lattice of fuel in water. Do you have any good experimental checks for the reactivity of a lattice with a geometry that is similar to the storage rack with fuel assemblies in it?

17. In your calculational method how is the inherent forward scattering of neutrons by hydrogen atoms accounted for in the water gap which surrounds the fuel assembly.

18. Please state the maximum U-235 loading in grams of U-235 per average axial centimeter of fuel assembly that you propose to put into this storage facility and compare this number to the one you used in your criticality calculations. Also, please state how you will know that this limit will not be exceeded during the life of this storage rack.

19. Please provide the sensitivity of the calculation k_∞ to the following changes (at least one point above and one below the nominal values):

    (a) Lattice pitch or thickness of water between fuel assemblies in the storage lattice.

    (b) The fuel loading in grams of U-235 per average axial centimeter of fuel assembly.

20. To show the absolute limit of reactivity please calculate the k_∞ of a cell of the proposed storage lattice with the most reactive fuel assembly in it as a function of the density of the water in the intercell space. Plot a curve of the k_∞ all the way from almost zero density H_2O in the intercell space to a density of 1.0 gm/cm^3. For these calculations hold the fuel assembly portion of the cell and its associated pure water at 68°F.

21. When considering the direct gamma heating from the highest power fuel assemblies into the stainless steel boxes and into the intercell water between the fuel assembly boxes in addition to the conducted heat please calculate the maximum temperature of the intercell water and the concomitant amount of natural circulation flow that gives this maximum temperature.

22. How much flow area will be provided for intercell water flow at the base of the racks; i.e., at the inlet to the intercell space?

23. If the fuel assembly lead ins at the top of the stainless steel boxes tend to close off the natural circulation of the intercell water, how much flow area will be provided for the natural circulation flow of the intercell water? Will this flow area be in a form such that it will be possible for it to be closed off by the buildup of crud, by bending the lead ins, or in some other way such that hot water and steam could possibly get trapped in the intercell spaces?

24. It appears that because of their larger temperature defect, assemblies fueled with the mixed oxide (UO_2 + PuO_2) may be more reactive at fuel pool temperature. If during the lifetime of this storage facility you forsee the possible use of mixed oxide fuel assemblies please either make sufficient allowance for it in your subcriticality or make the commitment that at that time you will remodify this facility if it does not meet the NRC's subcriticality requirement.

25. Consider the gamma heating of the fuel storage cell walls and intercell water and show that in the case of loss of all cooling systems, the reactivity of the facility will remain at a safe level.

26. State how the quality assurance measures for manufacturing these racks will ensure that the proper bundle spacing will be achieved and the amount of stainless steel required between fuel bundles will be correct. Indicate what on-site checks are made after assembly of the racks.

27. State how the temperature of the water in the fuel storage pool is monitored. Show the relationship between indicated temperature and maximum bulk water temperature in the pool. Give the alarm set point and the indicated temperature at which additional cooling measures would be initiated. Show the maximum temperatures that would be expected in the fuel pool.

28. To ensure that the procedural controls referred to in paragraph 5.8.3 of Amendment 78 are established to prevent the fuel pool water temperature from exceeding 140°F, please propose appropriate technical specifications and give their bases.

29. State the number of fuel assemblies now stored in the spent fuel pool, i.e., after refueling in preparation for cycle 6.

30. What is the average volume of water in the spent fuel pool (SFP)?

31. State the size of equipment in the purification system (volume and type of resin, volume of disposable part of filter) and the criteria for the replacement of the equipment.

32. Describe the normal flow for the purification system?

33. Discuss the frequency of operation for the present SFP and what frequency of operation is expected for the modified SFP.

34. State the amount and relate the present annual quantity of solid radwaste generated by the purification system to the expected increase in solid radwaste which will result from modifying the SFP.

35. Provide measured data regarding the release of Krypton-85, Tritium and Iodine-131 from the Fuel Building. If measured data are not available from the Fuel Building, provide this data for the overall ventilation which includes the Fuel Building.

36. What is the average burnup of the fuel in MWD/MT at present? What is the expected average burnup when the number of spent fuel assemblies in the pool reaches a maximum? When will this occur? How many fuel assemblies will be in the storage pool at that time?

37. Provide an analyses of the ESF ventilation filter assemblies for the fuel handling and cask tip accidents with respect to the positions in Section C of Regulatory Guide 1.52.

38. Provide an estimate of the increase of long-lived radionuclides released from the Fuel Building due to the modification of the SFP. Provide an estimate of the increase in the whole body and skin dose at the site boundary.

39. Provide (1) the number of bundles that could be struck by a cask fall or tip, including effects of any superstructure on the cask; (2) a conservative analysis of fission product release from fuel bundles potentially subject to impact assuming that the most recently off-loaded fuel is in the impact area; (3) a realistic (best estimate) radiological analysis of a cask fall or tip; and (4) any technical specifications proposed on the decay time required prior to loading storage positions within the zone which could be struck by a cask fall or tip.

40. Discuss the overhead cask handling system from the points of view of (1) yoke and/or cable failure, and (2) braking devices, their capacity and effect on the ability of the handling system to withstand possible sudden decelerations induced by rapid braking following a loss of power to the system.

Discuss all typical loads that may be carried near or over the spent fuel pool.

41. Table VII-1 does not show activity from crud radionuclides such as ^{58}Co and ^{54}Mn. Please show the isotopic analysis of these nuclides.

42. Based on the radionuclide concentrations in the spent fuel pool as shown in Table VII-1, the dose rate above the pool should be <0.5 mrem/hr. Please identify the sources of radioactivity that provide the dose rates that are greatly in excess of this value as shown in Figure 12. Explain why the LPRM's and channel clips, when stored in the fuel pool, provide comparatively excessive dose rates when compared to the stored spent fuel elements.

43. It would appear, from Figure 12, that the radiation levels at the edge of the pool would be due to crud (e.g., ^{58}Co, ^{60}Co). Unless this crud is removed, expansion of the spent fuel pool will provide additional crud build-up causing a further increase in dose rate levels. Please verify that you considered this build-up insofar as operational difficulties are concerned and describe any plans you may have for crud removal and the removal methods that will be used to reduce radiation levels at the sides of the pool to as low as reasonably achievable.

44. Please provide an estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter cartridges resulting from the fuel pool storage expansion.

45. Please specify the expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (11) and (12) above.

46. Please discuss the radiological dose impact that may be caused by any radionuclide that may be in the air above the pool including ^{131}I and ^{3}H.

5 ... -953

Exhibit 6

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

General Public Utilities Corporation

August 11, 1976
EA-76-779

Mr. George Lear, Chief
Division of Operating Reactors
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Lear:

SUBJECT: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Supplement No. 1 to Facility Description and
Safety Analysis Report Amendment No. 78

Pursuant to Title 10, Code of Federal Regulations, Section 50.59, three signed originals and fifty-seven conformed copies of Jersey Central Power & Light Company's Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report are herein submitted.

This Supplement responds to your letter of June 24, 1976, which presented questions on our proposed spent fuel pool modification submitted in Amendment No. 78 to the Facility Description and Safety Analysis Report. Amendment No. 78 was dated March 18, 1976. The analysis required for response to questions 5, 7, and 10 of your June 24 letter is not complete. Responses to these questions will be submitted within thirty days.

Very truly yours,

8169

Ivan R. Finfrock, Jr.
Vice President

dj
enclosure

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Application for Reactor Operating License

Supplement No. 1

to

Amendment No. 78

Docket No. 50-219

Applicant submits by this Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report responses to questions from the Nuclear Regulatory Commission on the proposed spent fuel storage pool modification.

JERSEY CENTRAL POWER & LIGHT COMPANY

BY ______________________
Ivan R. Finfrock, Jr.
Vice President

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed to before me this 11th day of August, 1976.

____
Notary Public

PHYLLIS A. KASIS
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Aug. 16, 1979

UNITED STATES OF AMER'CA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY)

CERTIFICATE OF SERVICE

This is to certify that a copy of Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station, dated August 11, 1976, and filed with the United States Nuclear Regulatory Commission on August 11, 1976, has this 11th day of August , 1976, been served on the Mayor of Lacey Township, Ocean County, New Jersey, by deposit in the United States mail, addressed as follows:

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

BY ______________________
Ivan R. Finfrock, Jr.
Vice President

DATED: August 11, 1976

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

General Public Utilities Corporation

August 11, 1976

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Scanlon:

Enclosed is one (1) copy of Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station.

This Supplement No. 1 to Amendment 78 was filed with the United States Nuclear Regulatory Commission on August 11, 1976.

Very truly yours,

Ivan R. Finfrock, Jr.
Vice President

dj
enclosure

Question 1

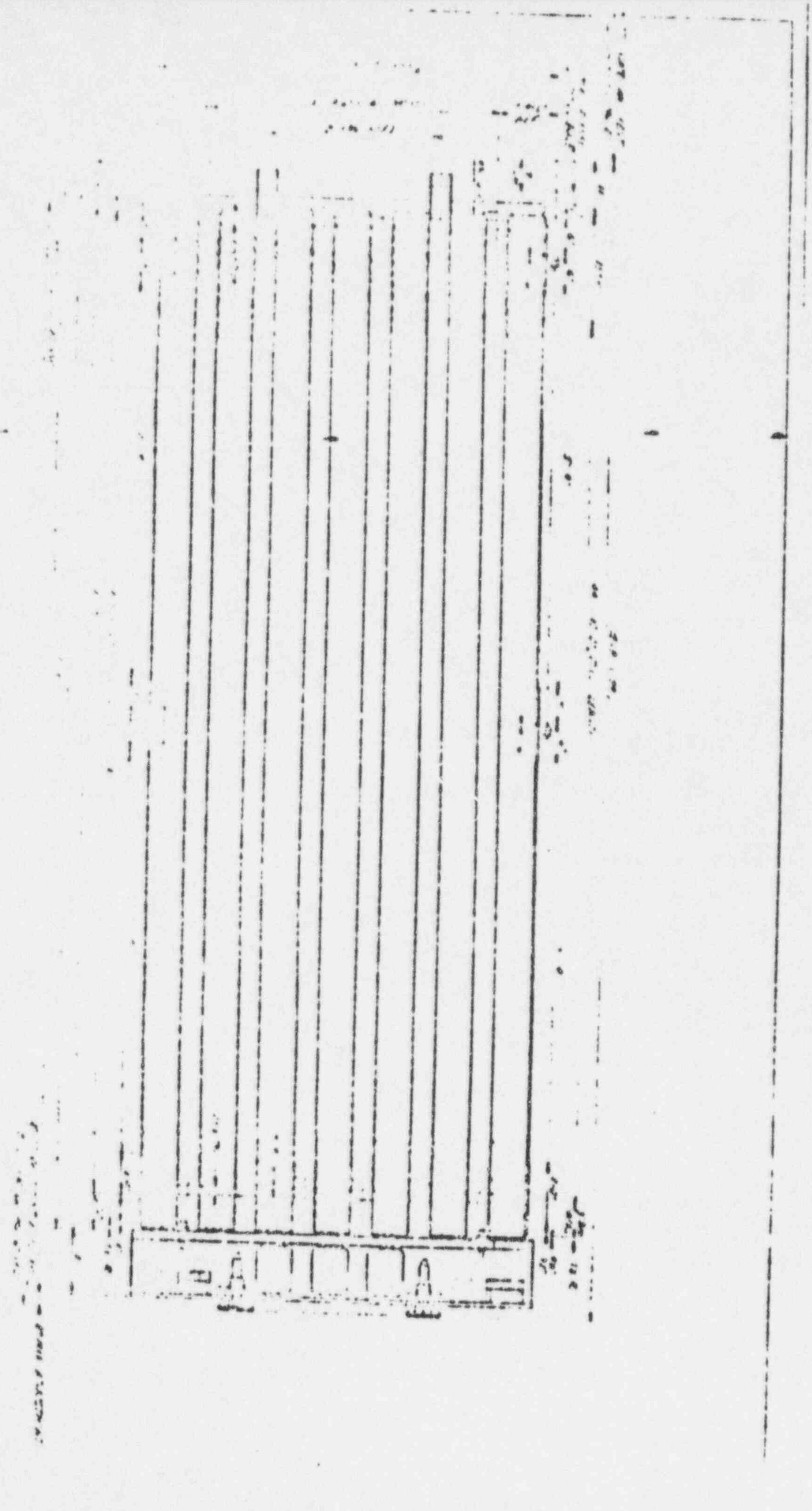
Provide sketches of the fuel pool storage racks which define the primary structural aspects and elements relied upon for the structure to perform its safety function. Include typical details of all interfaces with the pool boundaries and connections between rack assemblies. Also provide sketches of the fuel storage pool showing its principal dimensions and structural features and its relationship with surrounding end supports. Include sketches of the existing "swing bolts" and illustrate how racks will be placed in areas where there are no swing bolts.

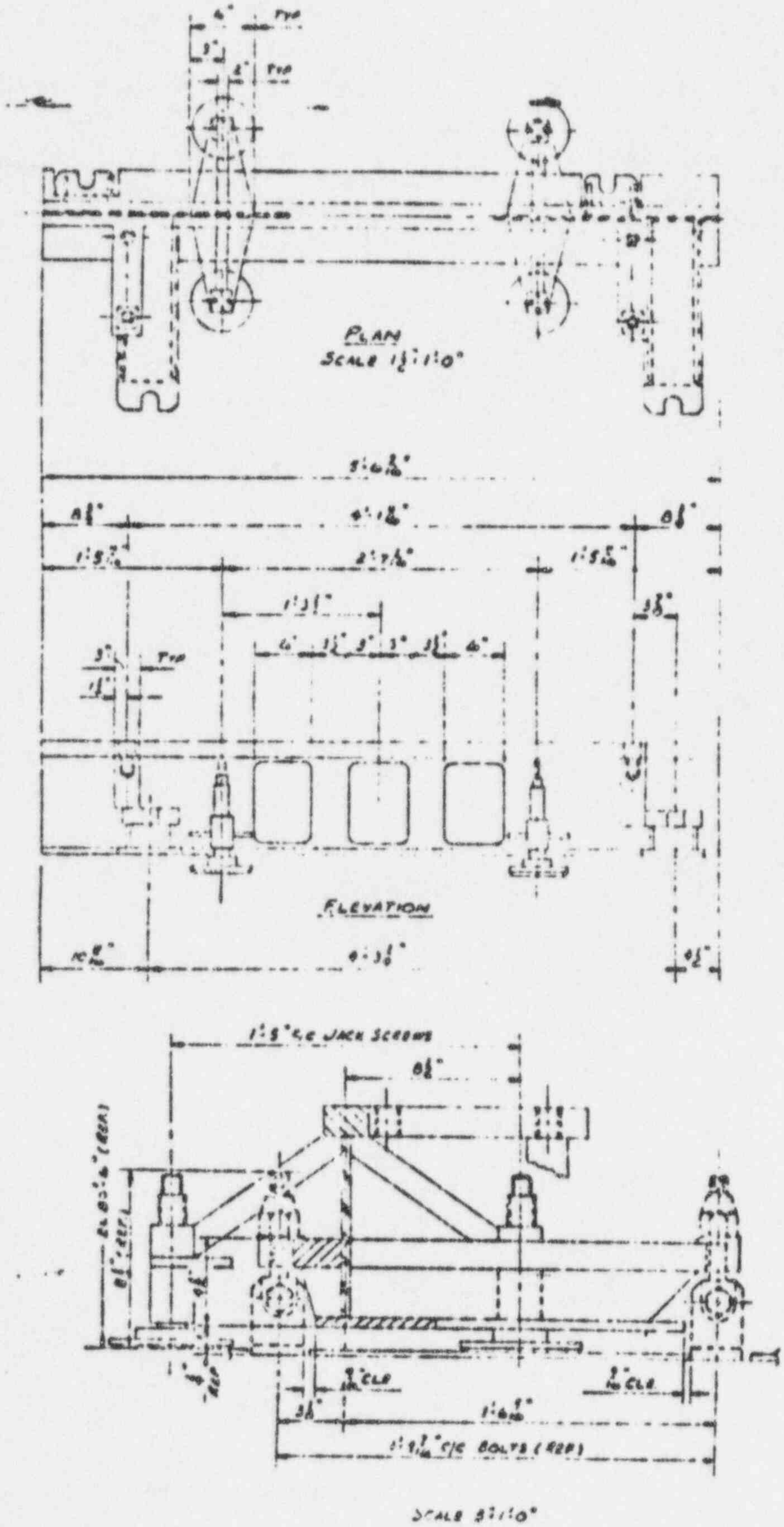
Answer

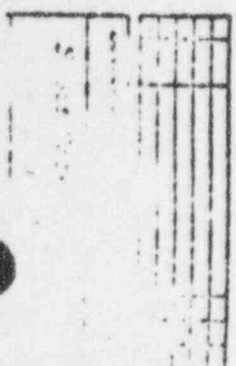
Fig. 1-1 is a sketch of a 28 cell fuel storage rack and base assembly, showing upper tie plates and rack hold down bolts.

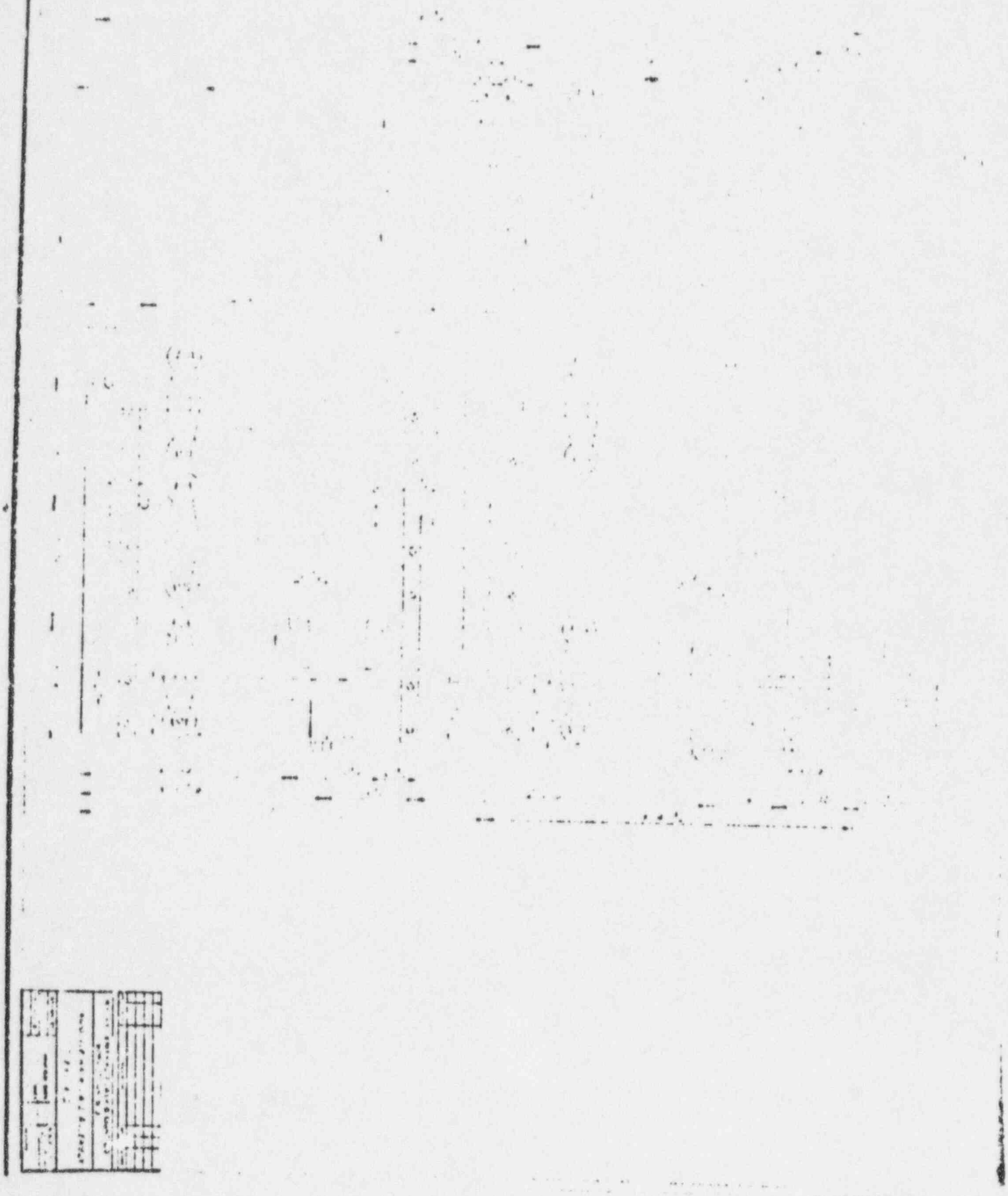
Fig. 1-2 shows a typical rack base with jackscrew leveling pads. Also shown are the connecting arms for the swingbolts.

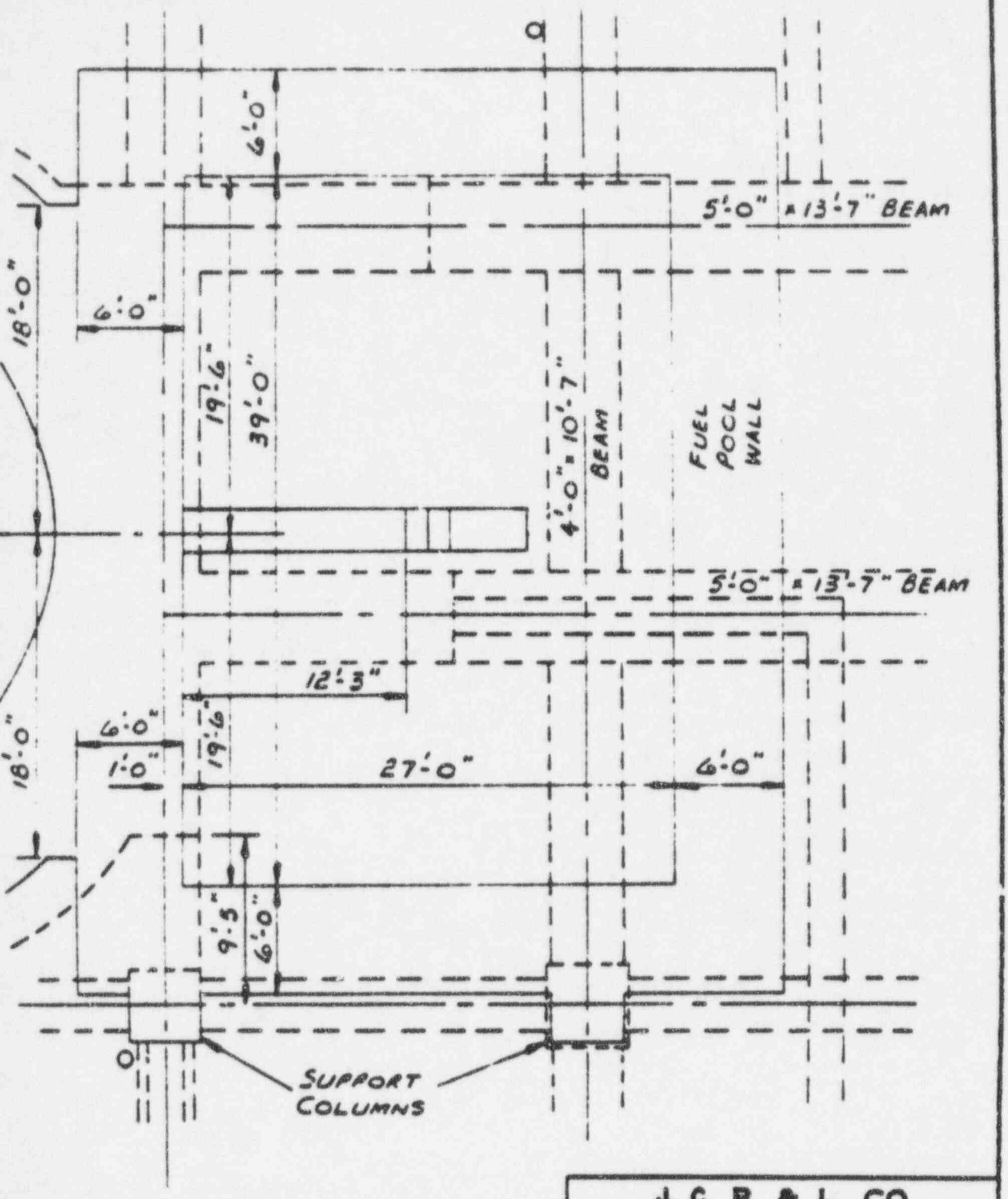
Composite bases, shown in Fig. 1-3 will be used to bridge areas where there are no swingbolts. The fuel pool support structure is shown in Figure 1-4.




PLAN
ELEVATION
C/C JACK SCREWS
C/C BOLTS (REF)







J. C. P. & L. CO.
MORRISTOWN, N. J.

SPENT FUEL POOL
SUPPORT BEAMS & COLUMNS
FIG. 1-4

SCALE 3/8" = 1'-0"
DRAWING NO. A

NO	DATE	REVISIONS	BY	CK	AUTHORIZED
1					
2					
3					
4					

DRAWN BY RJK 7-30-80
TRACED BY
CHECKED BY

Question 2

Provide sketches of the mathematical model of the fuel pool, fuel storage rack and fuel assembly system which will be used in the analysis and discuss the effects of submergence in water. Illustrate on the sketches the mechanism of shear and load transfer to the fuel pool walls and foundation slab. Discuss the possible impact of the fuel assemblies with the rack and describe how the effect of water sloshing in the fuel pool was analyzed.

Answer

A sketch of the fuel storage rack and base assemblies used to model a seismic event is shown on Figure 2-1. The analyses assumed east/west and north/south seismic events. The loads are transmitted through the base structure to the supporting jackscrews and swingbolt arm structures and ultimately to the spent fuel pool floor. No loads are transmitted to the fuel pool walls since there do not exist any structures connecting the racks to the walls. The east/west seismic event was analyzed using a single rack assembly (Figure 2-1). In this case the rack was assumed to pivot about a point in a counter-clockwise direction. The swingbouts and jackscrews were analyzed under this condition and found to be satisfactory.

In the case of a north/south seismic event, the analysis considered the mass of several racks moving as a single unit. The loads in this case were calculated to be less than the east/west seismic event, on both the swing bolts and jackscrews.

The fuel racks are designed to prevent a dropped fuel bundle from penetrating and occupying a position other than a normal fuel storage location. A dropped fuel bundle will probably end up in a final position that is somewhere between vertical and horizontal ontop of the racks. The only positive effect of such a bundle on the reactivity of the rack would be by virtue of a reduction in axial neutron leakage from the rack. Since the calculations reported here show the total axial neutron leakage effect to be only 0.0020 Δk, a dropped fuel bundle would not have any significant effect on the reported maximum possible reactivity of the spent fuel storage rack. The reactivity effect of a fuel bundle on the side of a rack has been calculated to be +0.0127. The worst case tolerance, temperature, etc. k_{oo} is still only 0.906 Δk.

The effect of wave sloshing against an end wall was analyzed by assuming a stationary vertical plate subjected to a horizontal water jet. From the momentum impulse equation the force generated by a 24 inch wave, provided that none of it washes out of the pool, will be negligibly small (force of 3701 lbs against a liner weight of 11,120 lbs). The effect of water sloshing does not appear to be significant when compared to the seismic loads on either the fuel racks or pool structures.

FIGURE 2-1

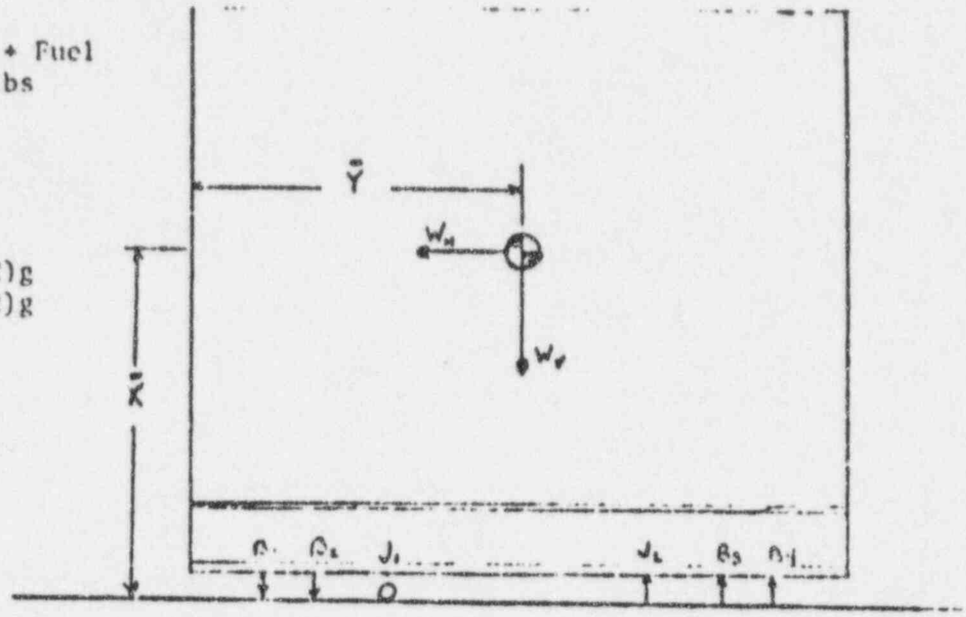
Single Rack Assembly - E/W Seismic Event

W_T of 32 cell rack + Base + Fuel + H_2O = 34,604 lbs

$\overline{X}$ = 85.44" $\overline{y}$ = 37.92"

W_H = Shear = 10,797 lbs

W_V = 45,400 lbs (1 + 0.312)g
= 23,808 lbs (1 - 0.312)g



Assume: CCW Rotation about point J_1

.˙. Jackscrew Force J_2 = 0 (compression only member)

& Swingbolts B_1, B_2 = 0 (tension only members)

Solving for loads on J_1 and B_3, B_4 by yields (worst case):

B_3 = 4,925 lbs

B_4 = 5,720 lbs

J = 17,230 lbs per screw at position J_1

Bolt Forces in Shear:

One B_s = 10,797 lbs

Two B_s = 5,399 lbs } Loads Equally Shared

Four B_s = 2,700 lbs

Question 3

Provide more specific information on the loads, load combinations and acceptance criteria which will be used in the design of the racks. Identify the magnitude of all loads considered in the design.

Answer

The new spent fuel racks were designed to withstand the most severe environmental, loading and seismic conditions which were assumed to occur simultaneously. The breakdown of the load combinations are as follows:

Normal plant operation:

1. Dead loads of racks, rack bases and seismic restraints.

2. Live loads of fuel assemblies: During all calculations, each rack was assumed to contain its maximum spent fuel capability.

3. Pool hydrostatic pressure load was neglected for rack structure calculations, since hydrostatic pressure has little effect on the open rack structure.

4. Pool water temperature was taken as ≤ 120°F.

5. Normal pool water circulation was assumed.

Severe environmental conditions: (includes normal operating loads). These loads included all the loads described above along with the following:

6. Trapped water effects treated as live loads within the rack.

7. Seismic accelerations (OBE) were 0.156g horizontal and (1 ± 0.156)g vertical; both components assumed to occur simultaneously in the "worst direction." The "worst direction" is defined as that direction which causes maximum bonding stresses of the welds at the bottom of the rack. The sign of (1 ± .156) depends on the specific component to be designed for worst case. Other seismic directions, both on-axis and off were checked to verify the "worst direction." The OBE acceleration (ground) component of 0.11 g was taken from FSAR Figure V-3-1. This value was scaled up to 0.156 g with the aid of Seismic Analysis of Reactor Building, Revised June 17, 1965, by John A. Blume and Associates, San Francisco, CA, for the pool floor elevation of 80' 6".

8. The effects of water sloshing were analyzed. The forces generated by water was much less than the seismic loads and was therefore neglected. (See Question 2).

Extreme environmental conditions: (includes normal operating loads). These loads include loads as previously described by items #1 through #6, #8, and the following:

9. Seismic accelerations (SSE) were 0.312 g horizontal and (1 ± .312) g vertical. As per the discussion in #7, both components were assumed to occur simultanciously in the worst direction. The SSE acceleration (ground) component was 0.22 g taken from FSAR, Section 3.1.1, page V-3-1 and -3-2. This value was scaled up to 0.312 g using the previously mentioned John A. Blume report.

Abnormal plant conditions: (includes some normal operating loads). These loads include #1 through #2 and #5 plus the following:

10. The spent fuel pool bulk temperature was assumed to be 200°F.

11. Thermal stresses due to sustained ΔT's across box walls, and depth of water were also considered, analyzed and found to be satisfactory and safe.

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 bath undergoing a safe shutdown earthquake. All important rack components were sized and stress-analyzed for the above "worst case" force loads and conditions. The racks and rack components were then analyzed for normal operating conditions, severe environmental conditions (OBE) and extreme environmental conditions (SSE), and were found satisfactory.

The new spent fuel racks are built to meet the stresses of Section VIII of the 1974 ASME Boiler and Pressure Vessel Code. Other design criteria have been examined and considered, but Section VIII was the most conservative (stress wise). During the design process, material properties of type 304 stainless steel, specifically the modulus of elasticity, E_y, and yield strength, σ_y, evaluated at 200°F were obtained from Appendix I of Section III of the B&PV Code since these properties did not exist in Section VIII. Appendix G of Section VIII directly references the AISC Manual for Steel Construction as a source for good structural practice. The AISC manual, in turn, references the governing section of the AWS Structural Welding Code which pertains to the performance of structural welding. Throughout the entire design of the spent fuel racks, Section VIII was the governing code. All evaluations of the applicable acceptance criteria were performed employing the 200° value of yield stress, so that a safety factor was included in all calculations regardless of actual pool water temperature.

Question 4

Describe the design and analysis procedures for the fuel storage racks, including the expected behavior under load and the mechanism of load transfer to the foundation. Discuss the resistance of the racks to sliding and underturning. Computer programs should be referenced to permit identification with available published programs.

Answer:

A design for the new spent fuel racks was evolved from due consideration of nuclear requirements, load-carrying capability, spatial arrangement, manufacturing requirements and ease of design modeling and analysis see (Table II-1 of Amendment 78.) The design bases for vibration and stress limits were selected from widely available and accepted standards (USNRC Regulatory Guides, ACI Codes, ASME Codes, AISC Codes, and others). Since it was a desired design goal to limit all primary stresses in the worst load condition undergoing the most severe seismic event to less than yield stress, all allowable limits were taken from codes and standards and were conservative for seismic design. Due to the relative simplicity of the design, an adequate, conservative mathematical model was available which lent itself readily to straight-forward analysis. No computer programs were necessary since the model was uncomplicated and easily adaptable to hand calculation. All critical components, supports, bolts, fasteners, beams and especially structural welds were carefully analyzed to the allowable values. In those cases where a required size by analysis fell between two standard sizes, the larger of the standard sizes was selected for the component resulting in a lower stress level and a more conservative design. After the design had been thoroughly analyzed, a review and audit by a licensed professional engineer competent in the structural field was performed. A final design review examined all facets of the rack structure and revealed no problems or design faults.

During the analysis, the beneficial effects of Coulombic friction ($F = \mu N$) were completely ignored; consequently the fuel assemblies were free to rock within their respective fuel boxes. The analysis assumed motions for the worst case so that a seismic load distribution could be determined and the worst case stress levels could be calculated. In this manner, all critical rack structures were analyzed and were verified to be satisfactory. All seismic loads were transmitted by the F/A's to the box structures, by the box structures through welds to the rack hold-down bolts, by these bolts through the base I-beam structure to the supporting jack stands and swing bolt arm structures, and finally to the spent fuel pool floor. No friction was assumed between jackstand bearing plates and the pool liner. For this frictionless condition, all base structures were subjected to the worst case seismic event, stresses calculated, and component size verified.

The fuel racks were secured to the floor by swing bolts restraining the base structure. The rack and base structure could not slide due to the shear restraint of the swing bolts and base swing bolt arm structure. For the same reason, rack lift-off was not possible.

Question 5

Discuss the extent to which the fuel pool has been analyzed to verify its ability to withstand the increase in overall loading. Identify the loads and load combinations investigated and the acceptance criteria for concluding that the original structure is adequate.

Answer

Will be answered later.

Question 6

Identify all the materials to be used in the fabrication and construction of the fuel pool racks. Describe the extent to which you intend to comply with ANSI N45.2.5, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants".

Answer

The materials used in the fabrication and construction of the rack assemblies and rack base assemblies will be as per the following specifications:

Plate - SS Type 304, ASTM A-240

Bar - SS Type 304, ASTM A-240

Sheet - SS Type 304, ASTM A-240

Rivets - SS Type 304, ASTM A-193

Bolts - SS Type 304, ASTM A-193

Nuts - SS Type 304, ASTM A-194

Weld-Material - SS Type 308, ASME SFA 5.9

ANSI N45.2.5 was not invoked verbatim since the racks are not considered structural members of the fuel pool. However, the ASME Codes and AWS Procedures required under the JCP&L procurement and fabrication specification for the spent fuel rack assemblies are those required by ANSI N45.2.5. Quality Assurance requirements for installation, inspection and testing will be in accordance with the JCP&L Operational Quality Assurance Plan which invokes all applicable requirements of ANSI N45.2.5.

Question 7

Indicate whether appropriate damping values and combination of modes and spatial excitation will be in accordance with Regulatory Guides 1.61 and 1.92 respectively for the analysis of the fuel pool and the fuel storage rack seismic system. It is noted that Table 11.1 references R.G. 1.92; however, the discussion in the first paragraph of Section 4.0 is not consistent with the guide.

Answer

Will be answered later

Question 8

The discussion in the second paragraph of Section 4.0 implies that the building response at the elevation of the spent fuel pool floor was used to prescribe the seismic loading on the racks. However, the discussion on the bottom of Page 4.0.3 indicates that the floor response spectrum at this elevation was used. Provide a more detailed discussion to clarify the design approach.

Answer

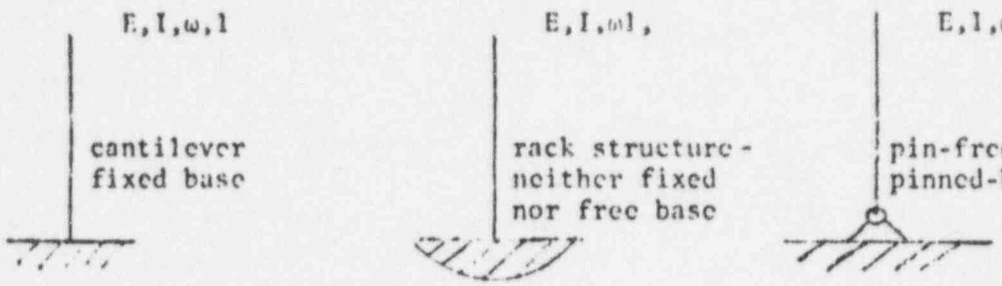
The revised Seismic Analysis of Reactor Building (By John A. Blume & Associates, San Francisco, CA June 17, 1965) was based on the seismic and geologic data as examined and reported by D. R. Housner for Jersey Central Power & Light's Oyster Creek Plant. This Blume Report determined the revised maximum accelerations acting on the reactor building due to D. R. Housner's recommended design earthquake. The most severe earthquake direction is assumed. The report presents the most credible maximum absolute accelerations for the building elevations due to the lateral seismic vibration, based on a ground motion of 0.11g. At various elevations in the building corresponding to Blume's lumped mass modeling, maximum accelerations are presented. Since the pool floor lies between two listed elevations, a linear interpolation based on building elevation was used to determine the most credible maximum absolute acceleration of the pool floor. The value for the pool floor was 0.156g based on 0.11g ground motion, and was used as the OBE horizontal and vertical seismic component. For the SSE ground motion of 0.22g, the pool floor acceleration was doubled to 0.312g. This value was used as the horizontal and vertical seismic component for all SSE rack analysis.

Question 9

It is indicated in the report that the fundamental frequency of the racks is greater than 33 Hz in both vertical and horizontal modes of vibration. Provide mode frequencies, mode shapes and participation factors for the first few modes to substantiate your position. The relevant dynamic model should also be presented. The staff position is that at the high frequency end, the ground response spectrum may not have any amplification over the maximum ground acceleration but contribution from significant modes should be considered for overall response

Answer

The racks for Jersey Central's Oyster Creek plant were modeled as cantilever beams with fixed bases (for horizontal vibration only). The rack structure can be bracketed by two models as sketched below:



From J. P. Den Hartog Mechanical Vibrations, 4th Ed, 1956, p. 432, the natural frequency of vibration for these structures is

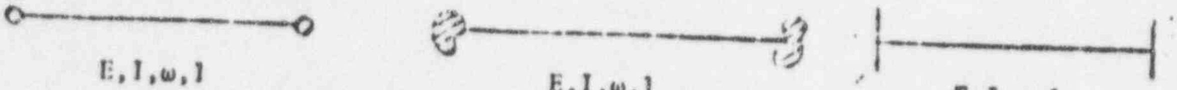
$$\omega_n = A_n \sqrt{\frac{EI}{\mu l^4}}$$

The coefficients A_N for the cantilever and the pin-free beam for the first three modes are:

Cantilever A_N	Pin-Free A_N
A_1=3.52	A_1=0 (no vibration mode)
A_2=22.0	A_2=15.4
A_3=61.7	A_3=50.0
	A_4=104

The most pessimistic model is the cantilever beam since it will result in the lowest natural frequency for our representative of the spent fuel rack horizontal vibrations.

The vertical vibration was modeled as pin-pin beams. Again, the rack structure can be bracketed by two models:



E,I,ω,l	E,I,ω,l	E,I,ω,l
Pin-Pin Beam Hinged ends	Rack Structure Neither hinged nor fixed ends	Clamp-Clamp Fixed ends

The coefficients for the first three modes are:

Pin-Pin A_N	Clamp-Clamp A_N
A_1=9.87	A_1=22.4
A_2=39.5	A_2=61.7
A_3=88.9	A_3=121.

The most conservative model is the pin-pin beam, as its natural frequencies are lowest for a given mode shape.

The following is the natural frequencies and mode shapes for the first 3 modes of the cantilever (horizontal) and pin-pin (vertical) vibration. The 4 x 2 rack structure is used as the example, and the frequencies calculated reflect the worst case values:

Mode	Cantilever Beam - Frequencies & Shapes
1	f1 = 38.65 Hz
2	f2=241.6 Hz
3	f3=677.5 Hz

Mode	Pin-Pin Beam - Frequencies & Shapes
1	f1 = 661.8 Hz
2	f2 = 2648. Hz
3	f3 = 5961. Hz

No higher modes are expected to contribute significantly to the overall response. This was confirmed by a model frequency analysis which also showed the static analysis of the structure to be quite satisfactory and conservative. Because the rack behaves as a solid metal structure, no softer substructures exist within the rack to generate lower modes and thus cross-coupling of the frequencies will not occur.

Question 10

Compare the most severe temperature distribution with that used in the original structural design of the fuel pool structure. Justify any increase in the maximum design basis transient temperature.

Answer

To be answered later

Question 11

Discuss the extent to which the behavior of each storage rack assembly was analyzed when fastened to other assemblies within the fuel pool. Discuss the effect on the rack design when only some of the assemblies are loaded with fuel.

Answer

The behavior of each storage rack was examined with respect to the connecting structures between the racks (see Figure 1 - for tie locations). Although Fig. 1 shows two tie bar elevations, analysis are currently under way to justify the elimination of the middle elevation tie bar without compromising safety, for the purpose of simplifying the installation. All welds, lugs, bolts, and attachments to the box walls were analyzed and were found satisfactory (all stresses less than yield). Since the mass ratio of the fuel assemblies to the total seismic mass (including deadweight and trapped water) is so large, no stress problems are created when the racks are only partially loaded. The mass percentages are nearly the same for either the 28 cell or 32 cell rack, and are: 13.1% deadweight steel, 21.6% trapped water, 61.3% fuel assemblies. The order of filling the rack with fuel assemblies is unimportant. The stresses are largest for the fully loaded rack condition.

Question 12

Discuss the ability of the rack to withstand the loadings imposed by a postulated dropped fuel assembly. Describe the loads and acceptance criteria and the design and analysis procedures utilized in the design. Include a discussion of the maximum drop height considered in the design, the masses involved, the kinetic energy at the point of impact and the amount of ductility utilized to dissipate the kinetic energy of the impact.

Answer

The ability of the new spent fuel racks to withstand the loads of a dropped fuel assembly is greater than 2 1/2 times that of the aluminum racks now in the pool. The new racks are stainless steel and considerally stronger than the aluminum racks. An analysis of maximum allowable force for constant given strain was performed. The governing factor is the modules of elasticity ($E_{ss}=28.3 \times 10^6$ psi, $E_{al} = 10 \times 10^6$ psi). Using a constant strain $\varepsilon = \sigma/E$) a ratio of the forces can be obtained:

$$\sigma = \frac{F}{A} = \quad \varepsilon = \frac{F}{AE}$$

For Constant E:

$$\left(\frac{F}{AE}\right)_{ss} = \varepsilon = \left(\frac{F}{AE}\right)_{al}$$

Where $A_{ss}/A_{al} = 0.922$. Thus the ratio F_{ss}/F_{al} is 2.61.

ε = constant strain

E = modules of elasticity

σ = stress

Question 13

Describe what controls will be exercised to ensure that the specified clearances between the fuel racks and the pool walls will be maintained during and after installation.

Answer

The specified clearance between the fuel racks and the pool walls will be maintained during installation by the use of approved procedures and after installation by the design of the fuel racks and the mounting to seismic I classification.

Question 14

The three inches stated for the width of the flux trap region on page 2.0-1 and the 5'-4" given for the length of the rack for 28 fuel bundles as given on page 10.0-2 are not consistent with the 9.650" storage lattice pitch given on the same page. Also, the 4.8285" half lattice pitch given on page 10.0-6 (Fig. 5) is not exactly consistent with the 9.650" lattice pitch given on page 10.0-2 (Fig. 1). Please provide a consistent set of numbers and drawings.

Answer

The correct width of the flux trap region is 3.602" (not 3" as stated on page 2.0-1). Six water boxes at this width plus seven fuel boxes each having a width of 6.055" results in a total length of 5"-4" (63.997"=5'4") for the 28 fuel bundle rack. The storage lattice pitch is the sum of 3.602" plus 6.055" or 9.657". The half lattice pitch is, therefore, 4.8285". The attached Figure 14-1 revises Figure 1, page 10.0-2, of Amendment 78.

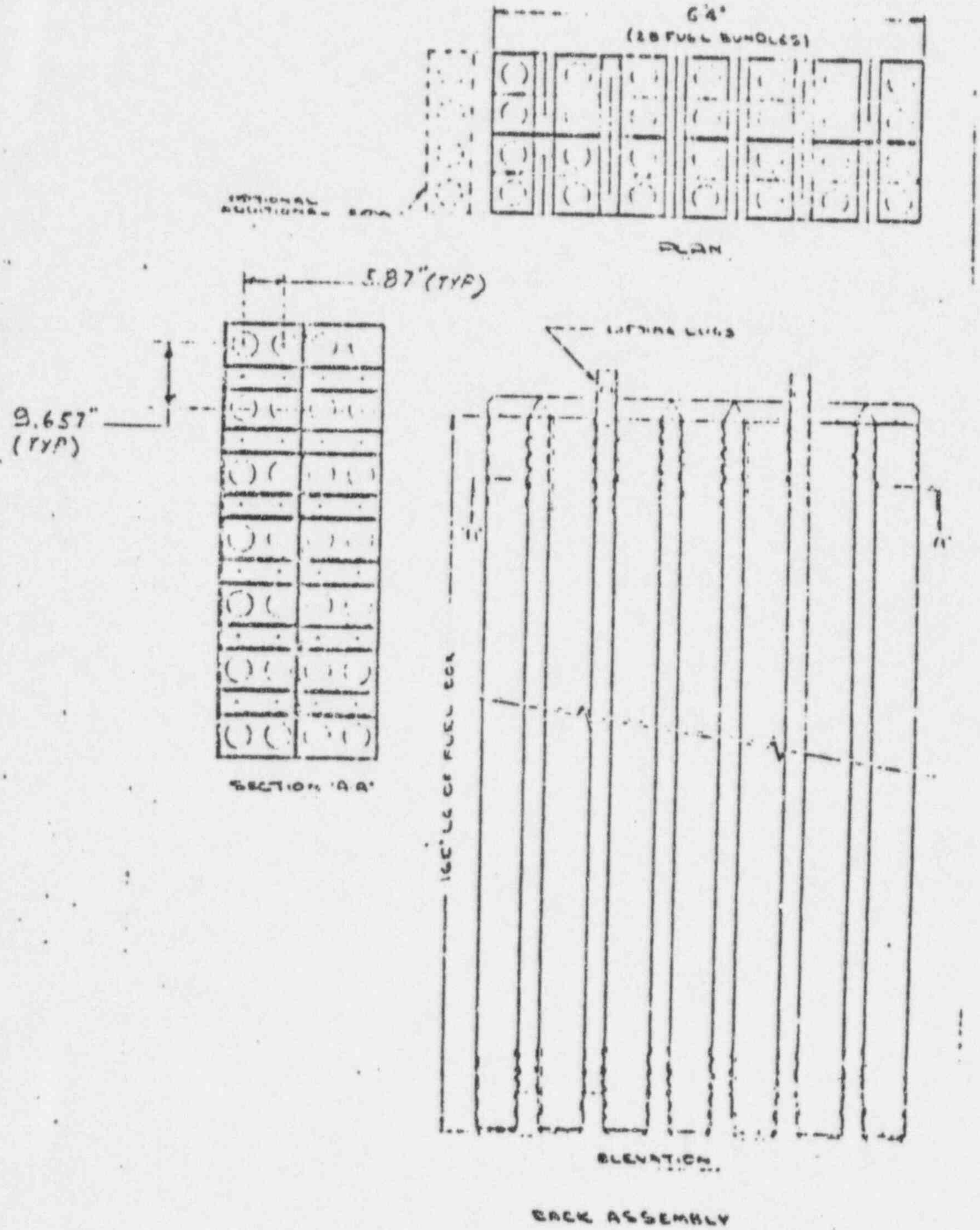

6'4"
(28 FUEL BUNDLES)
PLAN
5.87" (TYP)
9.657"
(TYP)
SECTION 'A-A'
LIFTING LUGS
ELEVATION
RACK ASSEMBLY

Figure 14-1

Question 15

In data submitted for reload cycle #5 the overall dimension of the type VB Fuel bundle is given as 5.217". But the overall fuel region size given in Figure 5 of FDSAR Amendment 78 is 5.166". What was the basis for choosing this 5.166" dimension for your calculations and what is the overall dimension of the maximum sized fuel bundle or assembly that will be put into this storage rack?

Answer

The fuel region size of 5.166 inches is based on a 7 x 7 array of fuel rods with a pitch of 0.738 inches. The VB fuel bundle tie plate, however, extends somewhat beyond the fuel rods and has an envelope dimension of 5.217 x 5.217 inches. The maximum sized fuel bundle which will be placed into this storage rack must fit inside a Zircaloy channel with an inside dimension of 5.278 inches.

Question 16

It appears that most of the critical experiments which you calculated consist of a uniform lattice of fuel in water. Do you have any good experimental checks for the reactivity of a lattice with a geometry that is similar to the storage rack with fuel assemblies in it?

Answer

We have been unable to locate any representative experiments with small arrays of partially enriched U-235 which are loosely coupled to one another. The experiments selected for validation of the model were small critical assemblies of partially enriched UO_2 rods surrounded by water reflectors. Since most of the neutrons which leave a fuel bundle in the storage facility are absorbed in the medium between the bundles, this validation is believed to be adequate, particularly in view of the sensitivity studies described in the answer to Question 17.

Question 17

In your calculational method how is the inherent forward scattering of neutrons by hydrogen atoms accounted for in the water gap which surrounds the fuel assembly?

Answer

Reactivity calculations are made with diffusion theory in which the solution of the Boltzmann Transport Equation for the neutron density is limited to the first 2 terms, and the Legendre expansion of the scattering cross section is also limited to 2 terms. The inherent forward scattering of neutrons is accounted for in the second term of the scattering cross section which includes the average cosine of the scattering angle. This angular dependence is reflected in the diffusion coefficient, D_i, in each neutron energy group in the PDQ calculation.

The sensitivity of results to the forward scattering effect has been investigated by varying the value of $B^2(\nabla^2\phi)$ used as input in the cross section generation code. This in turn causes significant changes in the group 1 diffusion coefficient. A change of the order of 15% in D_1, for the water gap region, which is considerably greater than anticipated errors between Diffusion and Transport theory, caused only a 0.4% change in lattice multiplication.

In the cross section generation code, the reference value of B^2 was near 0, which has been shown to yield better results in small critical assemblies where leakage effects are more important.

Question 18

Please state the maximum U-235 loading in grams of U-235 per average axial centimeter of fuel assembly that you propose to put into this storage facility and compare this number to the one you used in your criticality calculations. Also, please state how you will know that this limit will not be exceeded during the life of this storage rack.

Answer

The maximum U-235 bundle loading currently available is in bundles with a uranium loading of 485 pounds of UO_2 and an initial average U-235 enrichment of 2.63 w/o. This corresponds to a linear U-235 loading density of:

$$(485)(.0263)(453.6)\frac{238}{270} \div 366 = 13.9 \text{ gms/cm}$$

The calculations were performed with a bundle loading of 476 pounds of UO_2 with an average enrichment of 3.0 w/o U-235. This corresponds to 15.6 gms U-235/cm. This represents an upper limit since it is a higher fissile loading than is required to achieve the design burnup. It is also doubtful that fuel with enrichments much greater than 15.6 gm U-235/cm could be loaded into the reactor without violating control rod shutdown restrictions. This requirement is more restrictive than fuel storage facility criticality.

Question 19

Please provide the sensitivity of the calculated k_∞ to the following changes (at least one point above and one below the nominal values):

a) Lattice pitch or thickness of water between fuel assemblies in the storage lattice.

b) The fuel loading in gms of U-235 per average axial centimeter of fuel assembly.

Answer

Figure 19-1 shows the sensitivity of storage rack multiplication to spacing between fuel boxes. This figure assumes an infinite array of storage racks with no axial neutron leakage and a conservative value for the fast neutron diffusion coefficient in water.

Figure 19-2 shows the sensitivity of both fuel and storage rack multiplication to average fuel bundle enrichment. For both curves the assumptions are the same as those mode in determining Figure 19-1 values.

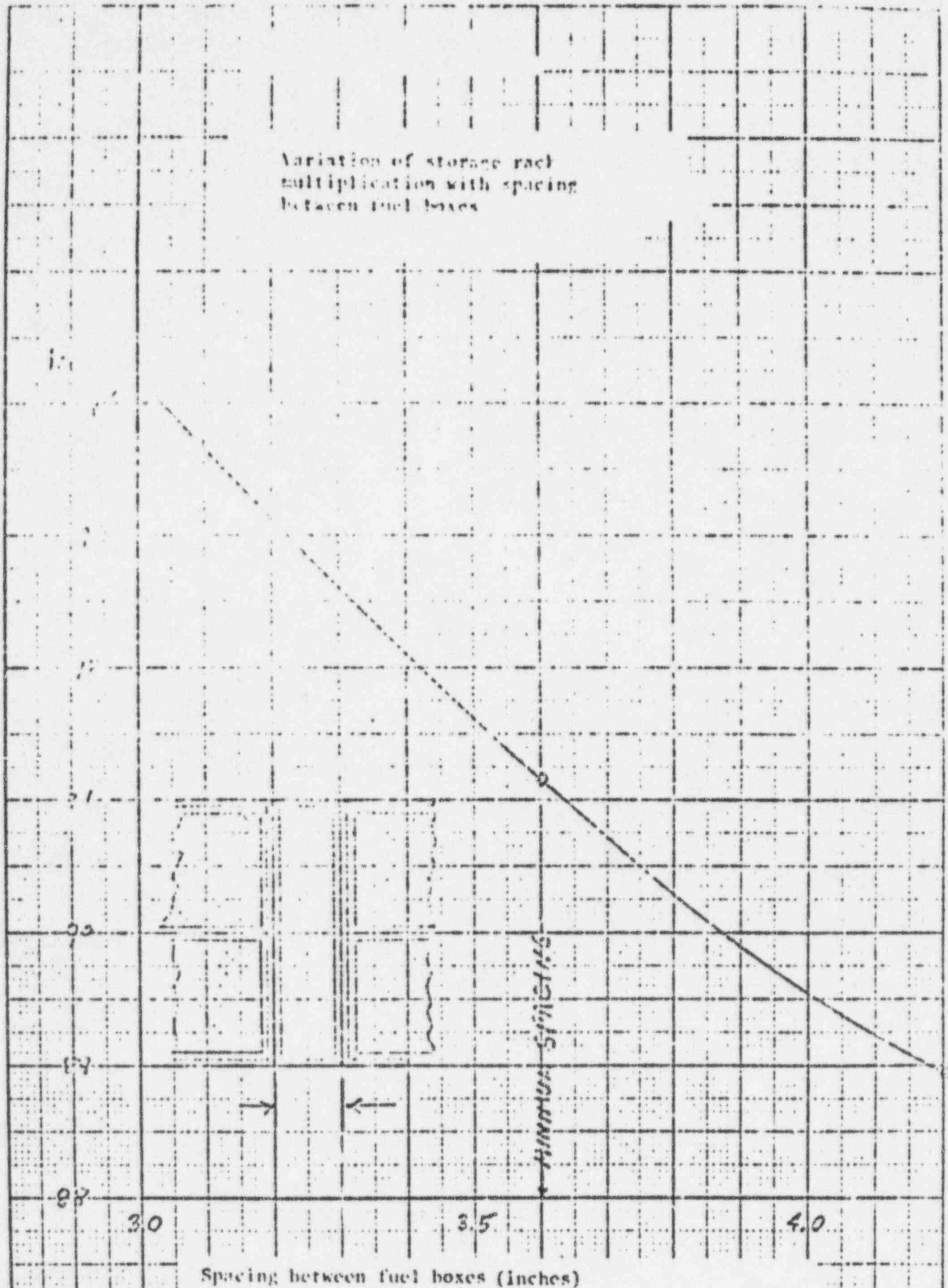

Variation of storage rack
multiplication with spacing
between fuel boxes
MINIMUM SPACING
3.0
3.5
4.0
Spacing between fuel boxes (inches)

FIGURE 19-1

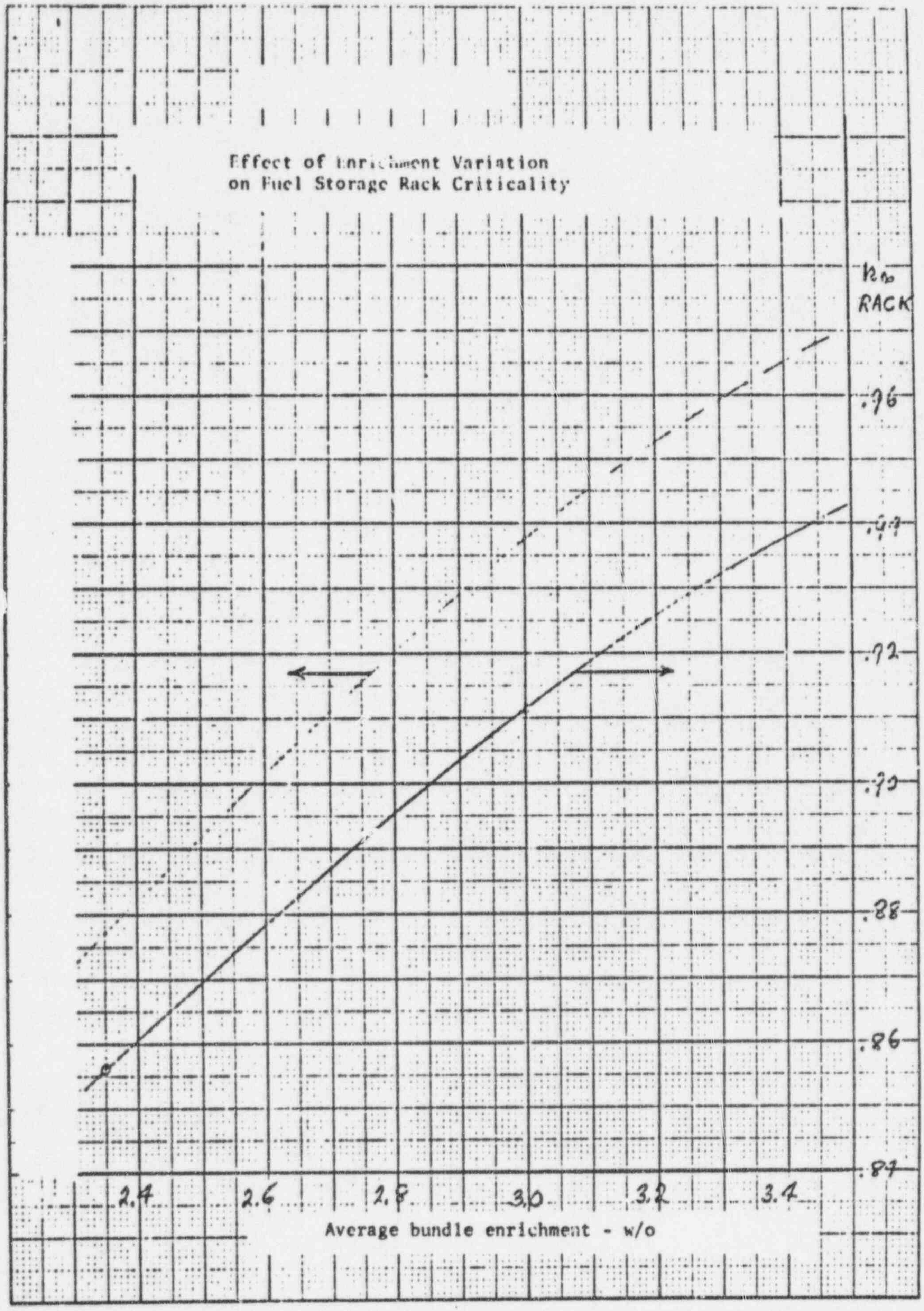

Effect of Enrichment Variation
on Fuel Storage Rack Criticality
k∞
RACK
.96
.94
.92
.90
.88
.86
.84
2.4
2.6
2.8
3.0
3.2
3.4
Average bundle enrichment - w/o

FIGURE 19-2

Question 20

To show the absolute limit of reactivity please calculate the k_∞ of a cell of the proposed storage lattice with the most reactive fuel assembly in it as a function of the density of the water in the intercell space. Plot a curve of the K_∞ all the way from almost zero density H_2O in the intercell space to a density of 1.0 gm/cm^3. For these calculations hold the fuel assembly portion of the cell and its associated pure water at 68°F.

Answer

Figure 20-1 shows the variation of storage rack multiplication with density of water in the "water box", i.e., the space between fuel storage boxes. This curve neglects both transverse and axial neutron leakage. Consideration of the latter will result in significant reductions in k_{eff} for lower water densities.

However, it should be noted that there is no reasonable postulated mechanism whereby the water density inside the boxes could be reduced without a corresponding or a larger reduction in the water density inside the fuel boxes where there are significant heat generation sources. A reduction in the water density inside the fuel boxes with a constant water density inside the water boxes will result in a sharp reduction in reactivity. Such a postulated change in water density represents a more realistic postulated abnormal situation.

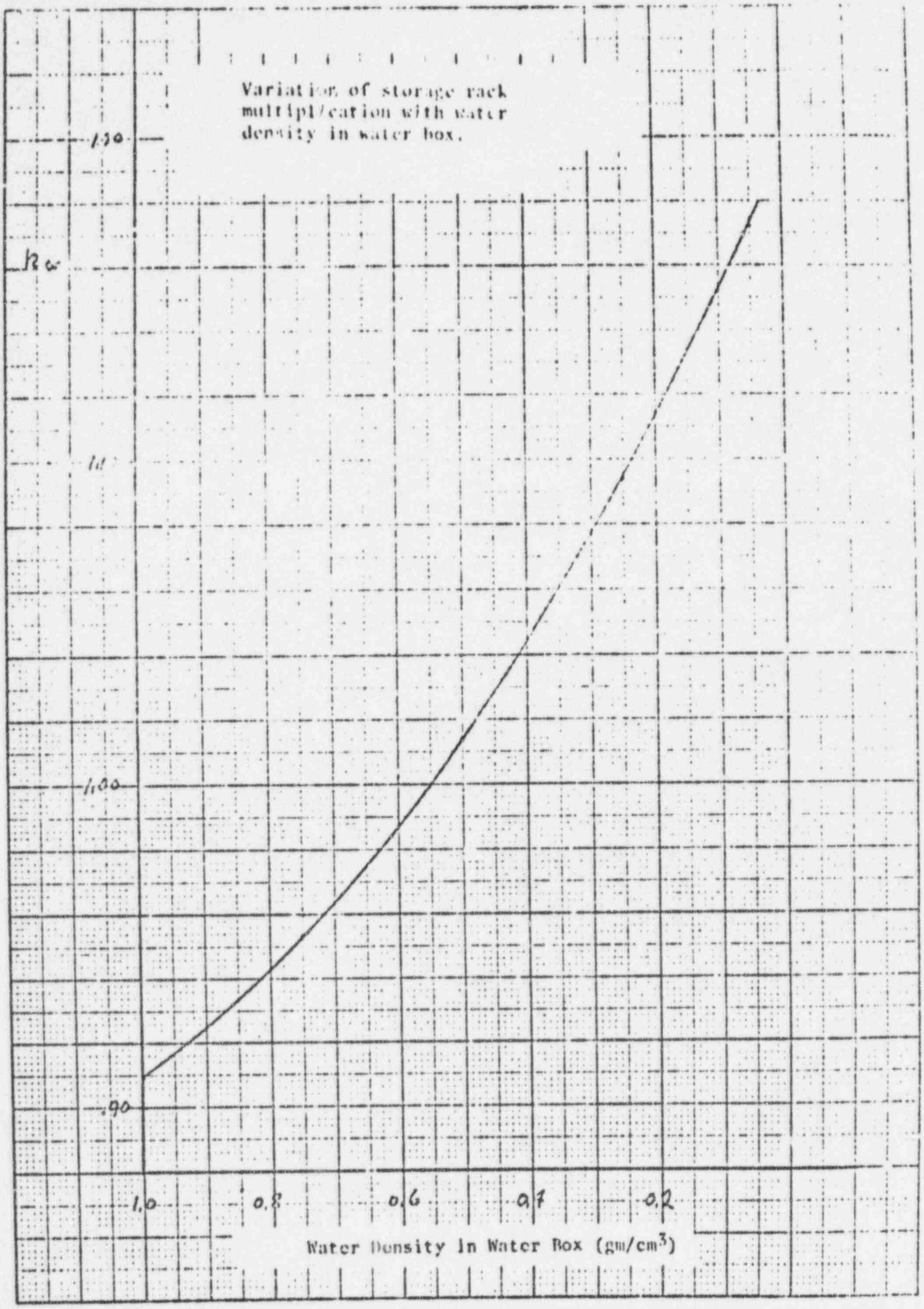

Variation of storage rack multiplication with water density in water box.
1.30
Rw
1.00
9
.90
1.0
0.8
0.6
0.4
0.2
Water Density in Water Box (gm/cm3)

FIGURE 20-1

Question 21

When considering the direct gamma heating from the highest power fuel assemblies into the stainless steel boxes and into the intercell water between the fuel assembly boxes in addition to the conducted heat please calculate the maximum temperature of the intercell water and the concomitant amount of natural circulation flow that gives this maximum temperature.

Answer

The direct gamma heating from the highest power fuel assembly has been analyzed within the following assumptions: all decay heating is assumed to be gamma; gamma flux is uniform and constant with assembly height; gamma absorption is proportional to material mass density; and material densities are assumed uniform.

Based upon a full core discharged at 11 days after reactor shutdown, with 120°F spent fuel pool heat exchanger discharge to the bottom of the pool, the temperature rise of the coolant in both the hottest fuel box and adjacent water box were calculated. The ΔT of the fuel assembly water is approximately 27°F whereas the ΔT of the water box is slightly less. The water box flow is 494 pounds per hour at these conditions.

Question 22

How much flow area will be provided for intercell water flow at the base of the racks i.e., at the inlet to the intercell space?

Answer

Intercell water flow at the base of the racks will be through two 3/4" diameter holes in the base plate of each water box.

Question 23

If the fuel assembly lead ins at the top of the stainless steel boxes tend to close off the natural circulation of the intercell water, how much flow area will be provided for the natural circulation flow of the intercell water? Will this flow area be in a form such that it will be possible for it to be closed off by the buildup of crud, by bending the lead ins, or in some other way such that hot water and stam could possibly get trapped in the intercell spaces?

Answer

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.

Question 24

It appears that because of their larger temperature defect, assemblies fueled with the mixed oxide (UO_2 + PuO_2) may be more reactive at fuel pool temperature. If during the lifetime of this storage facility you foresee the possible use of mixed oxide fuel assemblies please eigher make sufficient allowance for it in your subcriticality or make the commitment that at that time you will remodify this facility if it does not meet the NRC's subcriticality requirement.

Answer

If mixed oxide fuel is placed in spent fuel pool, appropriate reanalysis will be done to assure that the criticality criteria specified in Amendment 78 are met. If the criteria are not met, the necessary modifications will be made.

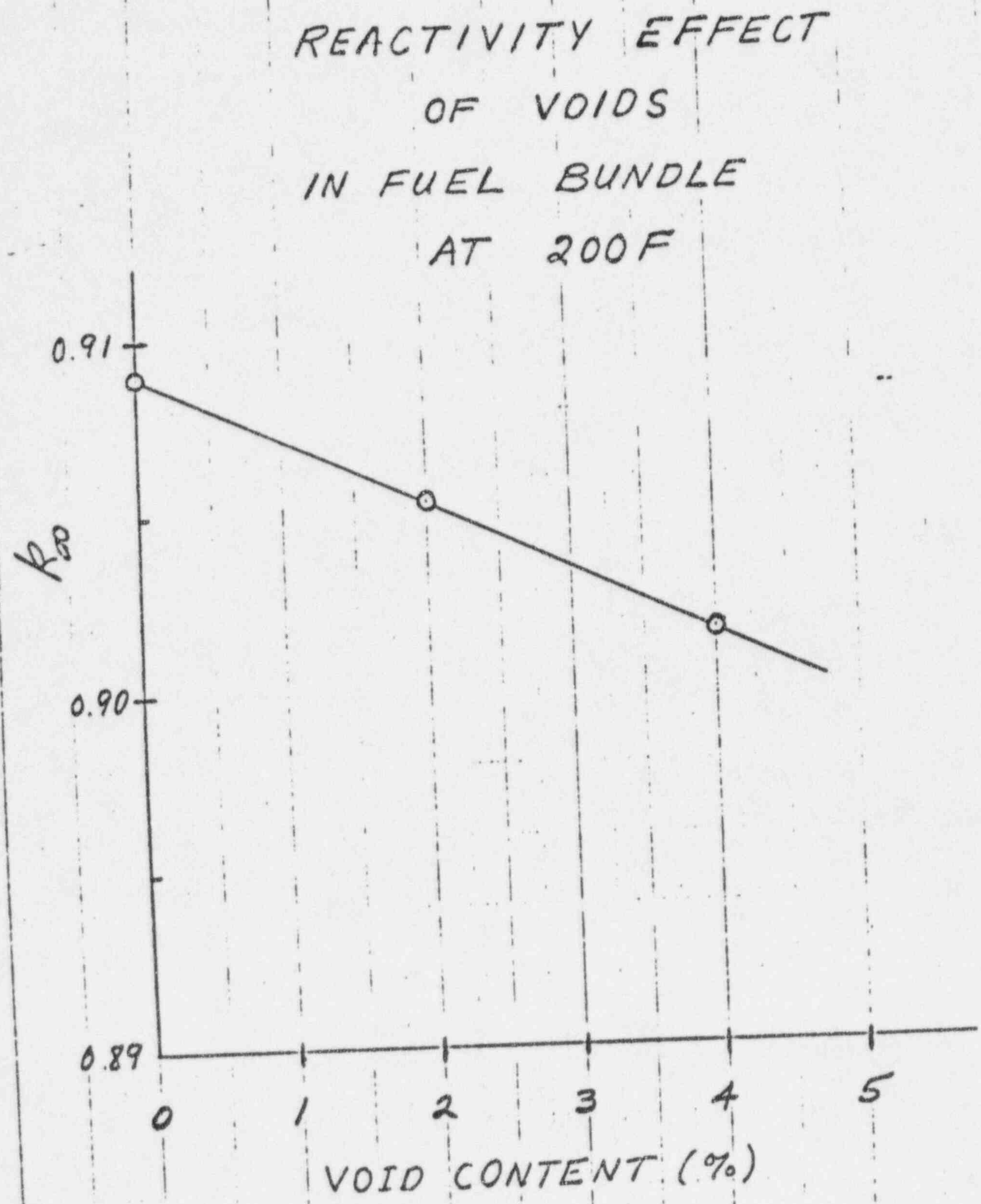
Question 25

In the case of a loss of all coolant systems consider a gamma heating of fuel cell and water cell walls and intercell water. Show that the reactivity will not increase.

Answer

The effects of introducing voids in the fuel bundles located in the spent fuel storage racks has been calculated. The attached Figure 25-1 shows that K_{∞} of the the rack array is reduced rather strongly as voids are introduced uniformly within the fuel bundle. The calculations employed a uniform moderator and a fuel temperature of 200°F.

A second set of calculations was also performed in which voids were also introduced in the water region surrounding the fuel bundle but inside the stainless steel box enclosing the fuel bundle. The behavior with void content was virtually identical since the maximum difference in the K_{∞} occurred at 4% voids and was only $0.00017\Delta k$.


REACTIVITY EFFECT
OF VOIDS
IN FUEL BUNDLE
AT 200F
0.91
0.90
0.89
k_∞
0
1
2
3
4
5
VOID CONTENT (%)

Question 26

State how the quality assurance measures for manufacturing these racks will ensure that the proper bundle spacing will be achieved and the amount of stainless steel required between fuel bundles will be correct. Indicate what on site checks are made after assembly of the racks.

Answer

The manufacturer of the racks for Jersey Central Power & Light (JCP&L) is an approved vendor according to the JCP&L Operational Quality Assurance Plan, which has been approved by the NRC. Dimensional checks at the manufacturer's facility will ensure proper spacing and the correct amount of stainless steel. The racks will be assembled and welded at the manufacturer's facility and will be visually inspected upon delivery to the Oyster Creek Station.

Question 27

State how the temperature of the water in the fuel storage pool is monitored. Show the relationship between indicated temperature and maximum bulk water temperature in the pool. Give the alarm set point and the indicated temperature at which additional cooling measures would be initiated. Show the maximum temperatures that would be expected in the fuel pool.

Answer

The fuel pool water temperature is currently monitored at the discharge of the spent fuel pool heat exchangers and indicated in the control room. A temperature indicator will be added to measure the pool surface temperature. Since the cooled water enters the pool at the bottom and the natural circulation forces the heated water to the surface, the surface temperature is an appropriate indicator of bulk temperature. This temperature indicator will be alarmed in the control room at $\leq 140°F$.

Question 28

To ensure that the procedural controls referred to in paragraph 5.8.3 of Amendment 78 are established to prevent the fuel pool water temperature from exceeding 140°F, please propose appropriate technical specifications and give their bases.

Answer

A procedure is being developed to specify the delay time before core off-load (see Section 5.5 of Amendment 78). This procedure will be based upon the analyses and references of Section 5 of Amendment 78 and will be completed and implemented before any core off-load. Such procedural control is effective, and the specification of spent fuel pool temperature limits in procedures, rather than the Technical Specifications, is consistent with the Commission's December, 1975, issue of the Standard Technical Specifications.

Note also that Question 10 will analyze the temperature distribution for the proposed pool modification. This will include the transient condition for pool bulk temperatures reaching 200°F.

Since the normal and transient temperature distributions will be analyzed to determine pool structural capabilities and since procedural control for pool bulk temperature will be instituted. additional control via Technical Specifications is unnecessary. Preservation of low fuel pool temperatures is also vital to allow a viable environment for personnel operations on the refueling floor. This is further motivation for proper procedure control.

Question 29

State the number of fuel assemblies now stored in the spent fuel pool (after cycle 6).

Answer

At present there are 326 fuel assemblies stored in the spent fuel pool.

Question 30

What is the average volume of water in the spent fuel pool (SFP)?

Answer

The average volume of water in the spent fuel pool is 40,000 cubic feet.

Question 31

State the size of equipment in the pruification system (volume and type of resin, volume of disposable part of filter) and the criteria for the replacement of the equipment.

Answer

The fuel pool demineralizer contain 150 cubic feet of the following resins:

cation - 100 cubic feet of strongly acidic cross-linked polystyrene, nalcite HCR-W,

anion - 50 cubic feet of Type I, strongly basic quarternary ammonium polystyrene nalcite ABR-P.

The demineralizer is instrumented with a conductivity monitor which alarms to the control room on high conductivity (>5.0 μmho).

The fuel pool filter consists of stainless steel wire-wound elements with a precoat of mixed resin. The filter is monitored for pressure and alarms to the control room on a high ΔP.

Question 32

Describe the normal flow for the purification system?

Answer

The Purification System consists of a filter and demineralizer, however operating experience shows that the fuel pool water quality can be maintained by use of the fuel pool filter alone. The normal flow through the Purification System is 400 gpm.

Question 33

Discuss the frequency of operation for the present SFP and what frequency of operation is expected for the modified SFP.

Answer

The filter is presently backwashed on a monthly basis, and this is not expected to change significantly with the modified spent fuel pool.

Question 34

Discuss the present annual quantity of solid radwaste generated by the purification system. Discuss the expected increase in solid radwaste which will result from modifying the SFP?

Answer

During the period from January 1, 1975 to December 31, 1975 a total of 34,319 cubic feet of solidified waste was shipped off-site in 162 shipments. The quantity directly accredited to the SFP purification system is not known since the waste from a number of filters is routed to a common hold up tank. The modified SFP is not expected to generate a significantly higher quantity of solid radwaste.

See also answer to Question 38.

Question 35

Provide measured data regarding the release of Krypton-85, tritium and Iodine-131 from the Fuel Building. If measured data are not available from the Fuel Building, provide this data for the overall ventilation which includes the Fuel Building.

Answer

Specific measurements for the Fuel Building are not available. However, measured data regarding the total stack release of Krypton-85, tritium, and Iodine-131, is available in the semi-annual operations report. The following table presents release data from the above isotopes during the period July 1, 1975 through December 31, 1975.

Isotope	Total release in Curies
Tritium *(Airborne)	.9663
Kr - 85_m (Gas)	2394.
I-131 (Particulate)	.0115
I-131 (Halogen)	1.4389

* Tritium measured in off gas and gland steam exhaust only

Question 36

What is the average burnup of the fuel in MWD/MT at present? What is the expected average burnup when the number of spent fuel assemblies in the pool reaches a maximum? When will this occur? How many fuel assemblies will be in the storage pool at that time?

Answer

The average burnup of the spent fuel in the spent fuel pool at present is about 20 MWD/MT. The average burnup is expected to increase in the future to a maximum batch average of 27 MWD/MT. This planned fuel pool modification will allow storage of ~1800 assemblies, which will permit storage of the spent fuel generated through about 1983 and still accommodate a full core unload as discussed in Section 1 of Amendment No. 78.

Question 37

Provide an analyses of the ESF ventilation filter assemblies for the fuel handling and cask tip accidents with respect to the positions in Section C of Regulatory Guide 1.52.

Answer

Section 11-52 of Supplement 6 to Amendment 68 of the Facility Description and Safety Analysis Report (FDSAR) presents a comparison to Regulatory Guide 1.52. Also, Jersey Central Power & Light Company's Technical Specification change proposal of December 17, 1975, proposed Limiting Conditions for Operation and Surveillance Requirements for the Stand-by Gas Treatment System consistent with the Commission's model Specifications. This proposal was incorporated into the Technical Specifications by Amendment No. 14, dated March 22, 1976. Section XIII-2.2 of Volume I of the FDSAR presents the analysis of the Refueling Accident and its fission product release. The cask tip accident is not considered due to the installation of a cask drop protection system at Oyster Creek which precludes the possibility of this type accident (See Amendment 68, Section 5.10 and Supplement No. 1 to Amendment 68, dated October 6, 1972).

Question 38

Provide an estimate of the increase of long-lived radionuclides released from the Fuel Building due to the modification of the SFP. Provide an estimate of the increase in the whole body and skin dose at the site boundary.

Answer

No increase in the release of long-lived radionuclides from the reactor building is anticipated. Recent examination of fuel assemblies removed from the reactor during refueling indicate that the fuel assemblies now being used have a high degree of integrity. The presence of fission product radionuclides in the pool water is due to leaking fuel rods. Most leaking fuel assemblies were from early fuel designs and most of the assemblies of these earlier design have been shipped off site. The new fuel now being used is not expected to exhibit significant leakage problems. Thus, the increase of long-lived radionuclides is expected to be insignificant. Note also that the increase in the inventroy in the spent fuel pool will be gradual. That is, only about 100 assemblies are replaced at each refueling. Therefore, an increase in radionuclide concentration will be detectable from the periodic pool water analysis.

Since no increase in the release of long-lived radionuclides is expected, whole body and skin dose at the site boundry is not expected to change.

Question 39

Assuming that pool integrity problems resulting from a cask drop will be resolved prior to cask movement, provide (1) the number of bundles that could be struck by a cask fall or tip, including effects of any superstructure on the cask: (2) a conservative analysis of fission product release from fuel bundles potentially subject to impact assuming that the most recently off-loaded fuel is in the impact; (3) a realistic (best estimate) radiological analysis of a cask fall or tip; and (4) any technical specifications proposed on the decay time required prior to loading storage positions within the zone which could be struck by a cask fall or tip.

Answer

A cask drop accident on or near stored fuel assemblies is not anticipated since the Oyster Creek spent fuel pool is equipped with a cask drop protection system (CDPS), as discussed in Amendment 68 to the Oyster Creek FDSAR, Section 5.10.* The cask will be moved from the equipment hatch to the CDPS at the northeast corner of the fuel pool and will not be moved over the fuel storage area at any time. Thus, any damage resulting from a cask drop accident is precluded.

* See also Supplement No. 1 to Amendment 68

Question 40

Discuss the overhead cask handling system from the points of view of (1) yoke and/or cable failure, and (2) braking devices, their capacity and effect on the ability of the handling system to withstand possible sudden decelerations induced by rapid braking following a loss of power to the system.

Discuss all typical loads that my be carried near or over the spent fuel pool.

Answer

Since the cask will not be moved over the fuel storage area, a yoke and/or cable failure is not expected to have any effect on stored assemblies. Either of these failures could result in the cask impacting a wall of the CDPS; however, the structural integrity of the liner would not be comprised. The cask handling system is discussed more fully in Amendment 68 to the Oyster Creek FDSAR, Section 5.10 and Supplement No. 1 to Amendment 68.

During normal operation loads over the spent fuel pool will be limited to spent fuel assemblies, weighing approximately 700 lbs.

Infrequent removal of waste can (200 lbs empty, 2250 lbs loaded) and cutting equipment (906 lbs) will occur. Analysis shows that there will be no structural damage to the pool or liner from a dropped waste can (loaded), and procedures restrict motion of this equipment to prevent it from being moved near spent fuel.

Question 41

Table VII-1 does not show activity from crud radionuclides such as ^{58} Co and 54 Mn. Please show the isotopic analysis of these nuclides.

Answer

Recent isotopic analysis of Cobalt-58 and Manganese-54 are as follows:

isotope	activity
Co^{58}	1×10^{-5} uc/ml
Mn^{54}	4.5×10^{-5} uc/ml

Question 42

Based on the radionuclide concentrations in the spent fueld pool as shown in Table VII-1, the dose rate above the pool should be 0.5 mrem/hr. Please identify the sources of radioactivity that provides the dose rates that are greatly in excess of this value as shown in Figure 12. Explain why the LPRM's and channel clips, when stored in the fuel pool, provide compariatively excessive dose rates when compared to the stored spent fuel elements.

Answer

The spent fuel pool is being utilized temporarily as a storage area for some high level radioactive waste. These materials are placed in buckets and hung from the sides of the spent fuel pool until ultimate desposition can be made. These sources increase the dose rates in the area surveys.

The LPRM's and channel clips provide comparatively higher dose rates than the fuel assemblies due to their higher elevation in the pool. The spent fuel assemblies are under twenty-three feet of water and the LPRM's and channel clips are under approximately five feet of water.

Question 43

It would appear, from Figure 12, that the radiation levels at the edge of the pool would be due to crud (e.g. ^{58}Co, ^{60}Co). Unless this crud is removed, expansion of the spent fuel pool will provide additional crud build-up causing a further increase in dose rate levels. Please verify that you considered this build-up insofar as operational difficulties are concerned and described any plans you may have for crud removal and the removal methods that will be used to reduce radiation levels at the sides of the pool to as low as reasonably achievable.

Answer

As stated previously, the higher radiation levels around the pool can be attributed to the waste material in temporary storage around the sides of the pool; however, the problem of crud build-up has been considered. Provisions have been made to vacuum clean the spent fuel pool prior to the installation of the new fuel storage racks. In the event of high radiation levels around the pool, the area will be roped off and procedures initiated to lower these levels. Various procedures, such as vacuum cleaning or removal of waste material, will be employed to reduce radiation levels if necessary.

Note again that the spent fuel pool inventory increase will be gradual (i.e. from periodic refuelings). Crud build-up will manifest itself in increasing radiation levels during the inventory increase should it become a problem. This gradual increase will allow detection of crud build-up and removal if necessary.

Question 44

Please provide an estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter cartridges resulting from the fuel pool storage expansion.

Answer

The annual man-rem burden from changing of resin and filter backwash is not expected to increase due to fuel pool expansion. The filter backwash operation is not expected to increase in frequency and will not result in additional exposure to personnel.

Question 45

Please specify the expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from 11 and 12 above.

Answer

No specific exposure data for the fuel pool area operations are available. However, since no significant increase in the pool water radionuclide concentrations or crud buildup is expected, no increase in the man-rem for the fuel pool area is expected.

Question 46

Please discuss the radiological dose impact that may be caused by any radionuclide that may be in the air above the pool including ^{131}I and ^{3}H.

Answer

Operating history has shown that radionuclides in the air above the pool have been insignificant. This is not expected to change with the proposed modification of the SFP.

Since the new fuel is not expected to leak, an increase in the iodine concentration is not expected. The operating temperature of the spent fuel pool will not be increased due to the modification, and therefore no significant increase in the release of iodine to the air over the pool is expected.

The tritium concentration is not a function of the number of fuel elements in the pool and therefore, the proposed modification will not alter the tritium concentration in this BWR facility.

Exhibit 7

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539-6111

General Public Utilities Corporation

EATJM-149
November 30, 1976

RECEIVED
DEC 2 1976
U.S. NUCLEAR REGULATORY COMMISSION

Mr. George Lear, Chief
Division of Operating Reactors, Branch #3
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Lear:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219 -1032
Request for Amendment to Provisional Operating License No. DPR-16 - Revision No. 1 to Technical Specification Change Request No. 44 and Addendum No. 1 to Supplement No. 1 to Amendment No. 78 of the Facility Description and Safety Analysis Report

Pursuant to Title 10, Code of Federal Regulations, Section 50.59, three signed originals and fifty-seven copies of Jersey Central Power & Light Company's request for Amendment to Appendix A of the Oyster Creek Nuclear Generating Station's Provisional Operating License No. DPR-16 and Amendment to the Facility Description and Safety Analysis Report are herein submitted. This request incorporates Revision No. 1 to Technical Specification Change Request No. 44, and Addendum No. 1 to Supplement No. 1, dated August 11, 1976, of Amendment No. 78 of the Facility Description and Safety Analysis Report (FDSAR). Amendment No. 78 and Technical Specification Change Request No. 44 were submitted to Mr. Stello on March 18, 1976, to incorporate the modifications and specifications necessary to accommodate the planned increase in storage capacity of the spent fuel storage pool of the Oyster Creek Station.

Supplement No. 1 to Amendment No. 78 of the FDSAR responded to the questions in your letter of June 24, 1976. Subsequent to this submittal several discussions were held between my staff and yours in which additional questions were raised about the planned spent fuel pool modification. The attached Addendum No. 1 to Supplement No. 1 and Revision No. 1 to Technical Specification No. 44 respond to those questions. All of the questions of your staff and your June 24 letter have been answered except those requiring analysis

12204

of the spent fuel pool structure (i.e. questions 5, 7 and 10 of your June 24 letter). That is, all questions about the planned increased capacity racks have been answered. Your staff has indicated that acceptable responses to these questions will permit the issuance of a license to install and utilize the racks but not increase the current capacity of the pool. This will facilitate the rack installation, which is scheduled to begin in early January, 1977, so that the schedule for the Spring, 1977, refueling outage (to begin in April, 1977) will not be jeopardized. It is understood that until the pool structural questions are resolved, no more than 840 spent fuel assemblies, the current limit, will be placed in the pool. It is anticipated that the pool structural questions will be resolved and submitted to you in early January, 1977.

The attached proposed Technical Specification Change has been reviewed and approved by the Plant Operations Review Committee and the General Office Review Board.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

cp

Attachment

Jersey Central Power & Light Company

MADISON AVENUE AT PUNCH BOWL ROAD • MORRISTOWN, N. J. 07960 • 201-539 6111

General Public Utilities Corporation

November 30, 1976

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Scanlon:

Enclosed is one copy of Revision No. 1 to Technical Specification Change Request No. 44 and one copy of Addendum No. 1 to Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station. These were filed with the United States Nuclear Regulatory Commission on November 30, 1976.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

cp

Enclosures

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Request for Amendment

Facility Description and Safety Analysis Report
Addendum No. 1
to Supplement No. 1
to Amendment No. 78

and

Revision No. 1 to
Technical Specification
Change Request No. 44
Docket No. 50-219

Applicant submits, by this Revision No. 1 to Technical Specification Change Request No. 44 to the Oyster Creek Nuclear Generating Station Technical Specifications, and by this Addendum No. 1 to Supplement No. 1 to Amendment No. 78 to the Oyster Creek Nuclear Generating Station's Facility Description and Safety Analysis Report responses to questions from the Nuclear Regulatory Commission on the proposed spent fuel storage pool modification.

JERSEY CENTRAL POWER & LIGHT COMPANY

By ______________________
Ivan R. Finfrock, Jr.
Vice President

STATE OF NEW JERSEY)

COUNTY OF MORRIS)

Sworn and subscribed to before me this ______ day of ______________, 1976.

____
Notary Public

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF)
) DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY)

CERTIFICATE OF SERVICE

This is to certify that a copy of Revision No. 1 to Technical Specification Change Request No. 44 and a copy of Addendum No. 1 to Supplement No. 1 to Amendment No. 78 to the Facility Description and Safety Analysis Report for the Oyster Creek Nuclear Generating Station, dated November 30, 1976, and filed with the United States Nuclear Regulatory Commission on November 30, 1976, has this 30th day of November, 1976, been served on the Mayor of Lacey Township, Ocean County, New Jersey, by deposit in the United States mail, addressed as follows:

The Honorable Edward J. Scanlon
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

By Ivan R. Finfrock
Ivan R. Finfrock, Jr.
Vice President

Dated: November 30, 1976

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION
(Docket No. 50-219)

PROVISIONAL OPERATING LICENSE NO. DPR-16

Applicant hereby requests the Commission to change Appendix A to the above-captioned license as follows:

1. Section to be Changed

   Section 5.3.B

2. Extent of Changes

   To specify the maximum k_{∞} of the spent fuel storage facility, to specify the maximum loading of U-235 for fuel to be stored in the spent fuel pool, to limit the weight of objects that can be moved over stored fuel in the spent fuel storage pool, and to limit the elevation of the spent fuel shipping cask above the top plate of the cask drop protection system.

3. Changes Requested

   Replace Page 5.3-1 with the attached Page 5.3-1.

4. Discussion

   Amendment No. 78, dated March 18, 1976 of the Oyster Creek Nuclear Generating Station's Facility Description and Safety Analysis Report (FDSAR) describes the modification of the spent fuel storage pool to expand its capacity by replacing existing storage racks with compacted storage racks.

   Supplement No. 1 to Amendment No. 78, dated August 11, 1976, responded to questions from the NRC on Amendment No. 78. Since the submittal of Supplement No. 1, discussion with the NRC Staff derived these proposed specifications. They have their bases in the analysis of the proposed spent fuel pool modification and in the analysis of the cask drop protection system.

   The changes and additions to the current Section 5.3 are denoted by a line in the right margin. Note that Specification 5.3.1.B was submitted with Amendment No. 78, but "K_{eff}" was specified rather than "K_{∞}". This change provides consistency with the FDSAR.

   Attached proposed Specifications 5.3.1, C, D, and E are added. As noted above they limit the U-235 loading to ensure a safe margin from criticality, and limit the weight of objects moved over stored fuel and the elevating of the cask above the cask drop protection system.

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 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.

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 $\leq$6 inches when it is above the top plate.

References

1. Amendment No. 78 to the Facility Description and Safety [illegible]ysis Report (Section 3)

2. Supplement No. 1 to Amendment No. 78 to the Facility De[illegible]ion 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 of the FDSAR (Ques[illegible] 12)

5. Supplement No. 1 to Amendment No. 78 of the FDSAR (Quest[illegible]n 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).

Instructions

Add the following pages to Supplement No. 1 to Amendment No. 78 of the FDSAR according to the page number.

QUESTION:

Provide details of the welding procedures used to fabricate each fuel box and the details and procedures used to fasten the boxes together in order to build up the honey-comb array.

ANSWER:

A typical 28 cell rack assembly is fabricated in the following manner:

Individual fuel and water boxes are formed by assembling two channel sections by weld joining with two continuous longitudinal fusion welds. Thirteen of the boxes (7 fuel/6 water) are welded to form the first row of the assembly. The boxes are joined one at a time with a series of TIG stitch welds 2" long, five per common side, on each of the two sides of the row. Each end of a box is then joined to the adjacent box with three 1" fusion stitch welds. The same procedure is followed to form the second row. After the second row is assembled, it is positioned on top of the first row and joined to its neighbors with three 1" fusion stitch welds. Additional row-to-row joining is achieved by TIG-arc spot welding at three intermediate levels along the length of the boxes. Exposed longitudinal joints between end or rows are welded with 2" TIG stitch welds in the same pattern as the longitudinal hidden joints. In all analyses the TIG-arc spot weld strength was conservatively modeled as a fusion plug weld (See figure 1-5 for weld locations).

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976

QUESTION:

Clarify how the racks are joined mechanically in minimum groups of eight. Figure 3 is not clear on this matter. In order to clarify the restraint provided, provide a sketch of one group of 12-23 element racks illustrated in Figure 14. Show the location of all the tie bars and the relationship of this group of racks with adjacent groups. Provide the minimum clearance to the fuel pool walls.

ANSWER:

The mechanical joining of racks is accomplished by the use of bolts and tie bars that fasten to lugs that are mounted at the top of each rack. These joints are shown in sketches Figure 1.6. The minimum clearance between racks and the pool wall is 11.3". Figure 1.6 depicts typically how a group of racks are joined together. However, as discussed in the addendum to Question 9, a minimum grouping of 12x2 will be used in lieu of the 4x2 array initially stated.

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976.

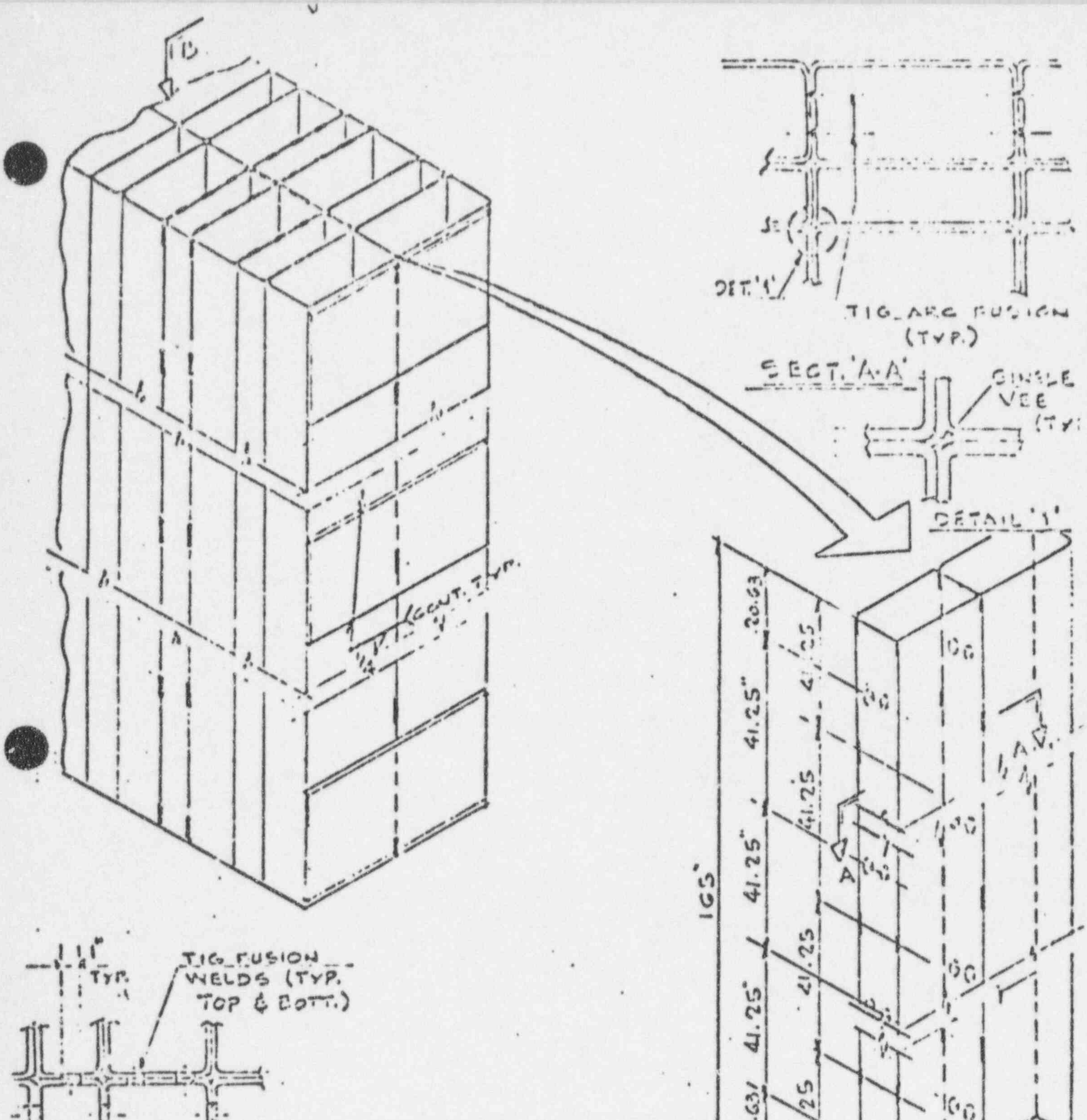
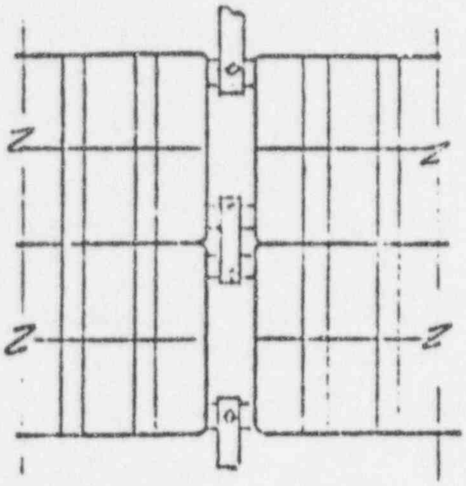
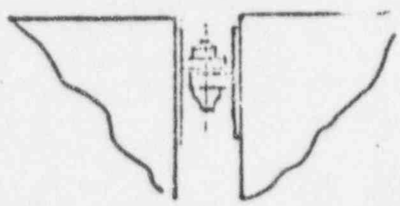


FIGURE 1-5 WELDING

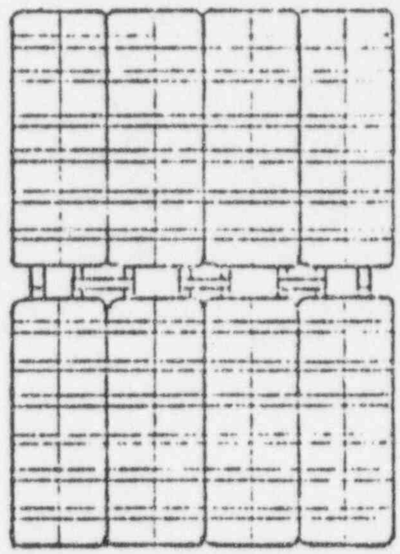
*Addendum 1 to Supplement 1 to Amendment 78 November, 197



TYPICAL PLAN



ELEVATION
TYP.
RACK MECHANICAL CONNECTION



PLAN OF GROUP OF
8 RACK ASSY'S

*Addendum 1 to Supplement 1 to Amendment 73 November, 1976.

Figure 1-6

QUESTION:

The discussion presented indicates that the E/W and N/S seismic events were considered independently. It is the staff's position that all three directions of earthquake motion be considered and combined in accordance with Regulatory Guide 1.92.

ANSWER:

The method used for representing and combining directional seismic responses is adequate since it can be shown that peak values of stress are in all cases identical to those obtained following the procedures recommended in Regulatory Guide 1.92.

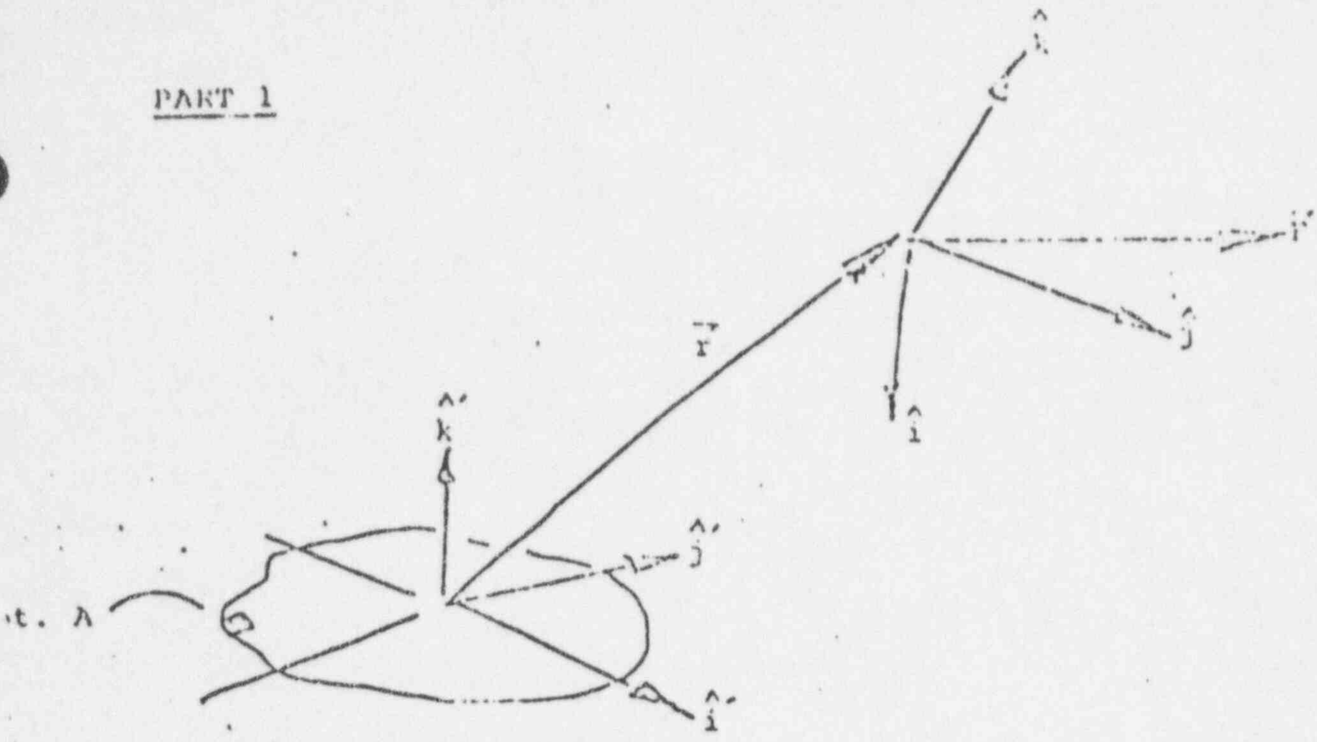
This method determines the direction by systematic search or by variational techniques, which yields the maximum effect or stress that would be calculated by combining the stress produced by seismic inputs in three mutually orthogonal directions, calculated independently, by the method of Square Root of the Sum of the Squares (SRSS).

The supporting analysis is presented in two parts. In part one, it is shown that the stress on an element in a structural cross-section due to a vector force applied at a point on that structure may be represented as a scalar (dot) product of the force vector dependent on the section properties and the spatial relationship of section and point of load application.

In part two it is shown that the maximum value of this scalar product is identical to the stress calculated by SRSS of the three principal orthogonal values of the force vector.

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976.

PART 1



7-1

The stress condition at pt. A on the cross section may be represented as a normal stress due to a force component in the $\hat{k}'$ direction, bending moments about the $\hat{i}'$ and $\hat{j}'$ axes and a shear stress due to shear force components in the $\hat{i}'$ and $\hat{j}'$ directions and torsion about the $\hat{k}'$ axis. For example:

$$\sigma_{normal} = \frac{1}{A}\vec{F}\cdot\hat{k}' + \frac{C_{AX}}{I_{XX}}\vec{r}\times\vec{F}\cdot\hat{i}' + \frac{C_{AY}}{I_{YY}}\vec{r}\times\vec{F}\cdot\hat{j}'$$

$$\tau = \frac{C_{SX}}{A_{SX}}\vec{F}\cdot\hat{i}' + \frac{C_{SY}}{A_{SY}}\vec{F}\cdot\hat{j}' + \frac{C_{AP}}{I_P}\vec{r}\times\vec{F}\cdot\hat{k}'$$

with C's, A's, I's appropriately chosen section constants. From vector laws:

$(\vec{a}\times\vec{b})\cdot\vec{c} = \vec{c}\cdot(\vec{a}\times\vec{b})$ and

$\vec{c}\cdot(\vec{a}\times\vec{b}) = \vec{b}\cdot(\vec{c}\times\vec{a}) = \vec{a}\cdot(\vec{b}\times\vec{c})$

then

$$\vec{r}\times\vec{F}\cdot\hat{i}' = \vec{F}\cdot(\hat{i}'\times\vec{r})$$
$$\vec{r}\times\vec{F}\cdot\hat{j}' = \vec{F}\cdot(\hat{j}'\times\vec{r})$$
$$\vec{r}\times\vec{F}\cdot\hat{k}' = \vec{F}\cdot(\hat{k}'\times\vec{r})$$

Then normal and shear stress may be represented in the following form:

$$\sigma_{normal} = \vec{F}\cdot\left[\frac{\hat{k}'}{A} + \frac{C_{AX}}{I_{XX}}(\hat{i}'\times\vec{r}) + \frac{C_{AY}}{I_{YY}}(\hat{j}'\times\vec{r})\right]$$

$$\tau = \vec{F}\cdot\left[\frac{C_{SX}}{A_{SX}}\hat{i}' + \frac{C_{SY}}{A_{SY}}\hat{j}' + \frac{C_{AP}}{I_P}(\hat{k}'\times\vec{r})\right]$$

The vectors on the right hand side [illegible] independent of the magnitude or di[illegible]on of $\vec{F}$.

define σnormal $= \vec{F}\cdot\vec{E}_A{}'$
τ $= \vec{F}\cdot\vec{T}_A{}'$

The vectors $\vec{E}_A{}'$, $\vec{T}_A{}'$ may be represented in the $\hat{i}$, $\hat{j}$, $\hat{k}$ system by the orthogonal transformation $[\phi]$ from $\hat{i}'$, $\hat{j}'$, $\hat{k}'$.

$$\vec{E}_A = [\phi]\vec{E}_A{}' = E_{AX}\hat{i} + E_{AY}\hat{j} + E_{AZ}\hat{k}$$

$$\vec{T}_A = [\phi]\vec{T}_A{}' = T_{AX}\hat{i} + T_{AY}\hat{j} + T_{AZ}\hat{k}$$

PART 2

Let $\hat{i}$, $\hat{j}$, $\hat{k}$ be the unit vectors in the directions in which the principal orthogonal seismic directions are specified and let F_X, F_Y, F_Z be the specified bounding values of that response. Then according to R.G. 1.92 the peak normal stress will be calculated as

$$\sigma peak_{1.92} = \sqrt{(E_{AX}F_X)^2 + (E_{AY}F_Y)^2 + (E_{AZ}F_Z)^2}$$

and the shear stress will be

$$\tau peak_{1.92} = \sqrt{(T_{AX}F_X)^2 + (T_{AY}F_Y)^2 + (T_{AZ}F_Z)^2}$$

Consider the generalized alternative method used for the calculation of peak stresses for the spent fuel storage racks - namely permit $\vec{F}$ to act in any direction with a magnitude bounded by an ellipsoid defined by principal radii Fx, Fy, Fz, and locate the peak value

$$\sigma max = (\vec{F}\cdot\vec{E}_A)max$$

$$\vec{F} = f_x\hat{i} + f_y\hat{j} + f_z\hat{k}$$

To find the stationary value of $\vec{F}\cdot\vec{E}_A$, set:

$$d(\vec{F}\cdot\vec{E}_A) = 0$$

$$(d\vec{F})\cdot\vec{E}_A + \vec{F}\cdot(d\vec{E}_A) = 0$$

Since $\vec{E}_A$ is invariant, $d\vec{E}_A = 0$ and

$$(d\vec{F})\cdot\vec{E}_A = 0 = E_{AX}\,df_x + E_{AY}\,df_y + E_{AZ}\,df_z$$

$$(d\vec{F}) = \hat{i}\,df_x + \hat{j}\,df_y + \hat{k}\,df_z$$

For the magnitude of $\vec{F}$ bounded by the ellipsoid whose principal radii are F_x, F_y, F_z,

$$\frac{f_x^2}{F_x^2} + \frac{f_y^2}{F_y^2} + \frac{f_z^2}{F_z^2} = 1$$

$$\frac{2f_x df_x}{F_x^2} + \frac{2f_y df_y}{F_y^2} + \frac{2f_z df_z}{F_z^2} = 0$$

Solve for df_z:

$$df_z = -\frac{F_z^2}{f_z}\left[\frac{f_x(df_x)}{F_x^2} + \frac{f_y(df_y)}{F_y^2}\right]$$

Substitute into $(d\vec{F})\cdot\vec{E}_A = 0$

$$E_{AX}df_x + E_{AY}df_y - E_{AZ}\frac{F_z^2 f_x}{F_x^2 f_z}df_x - E_{AZ}\frac{F_z^2 f_y}{F_y^2 f_z}df_y = 0$$

$$\left(E_{AX} - E_{AZ}\frac{F_z^2}{F_x^2}\frac{f_x}{f_z}\right)df_x + \left(E_{AY} - E_{AZ}\frac{F_z^2}{F_y^2}\frac{f_y}{f_z}\right)df_y = 0$$

For arbitrary df_x, df_y

$$E_{AX} - E_{AZ}\frac{F_z^2}{F_x^2}\frac{f_x}{f_z} = 0;$$

$$E_{AY} - E_{AZ}\frac{F_z^2}{F_y^2}\frac{f_y}{f_z} = 0;$$

$$\boxed{\begin{aligned} f_x &= \frac{E_{AX}}{E_{AZ}}\frac{F_x^2}{F_z^2} f_z \\ f_y &= \frac{E_{AY}}{E_{AZ}}\frac{F_y^2}{F_z^2} f_z \end{aligned}}$$

Solve ellipse equation for f_z

$$f_z = \sqrt{F_z^2 - \frac{F_z^2}{F_x^2}f_x^2 - \frac{F_z^2}{F_y^2}f_y^2}$$

$$f_z = \frac{E_{AZ}F_z^2}{\sqrt{E_{AX}^2F_x^2 + E_{AY}^2F_y^2 + E_{AZ}^2F_z^2}}$$

Then the stationary (maximum) stress occurs for the f_x, f_y, f_z thus calculated.

The value of that stress, then, is

$$\sigma_{max} = E_{AX}f_{x\,max} + E_{AY}f_{y\,max} + E_{AZ}f_{z\,max}$$

$$\sigma max = \frac{E_{AX}^2F_x^2 + E_{AY}^2F_y^2 + E_{AZ}^2F_z^2}{\sqrt{E_{AX}^2F_x^2 + E_{AY}^2F_y^2 + E_{AZ}^2F_z^2}}$$

$$\sigma max = \sqrt{E_{AX}^2F_x^2 + E_{AY}^2F_y^2 + E_{AZ}^2F_z^2}$$

An identical process is performed for shear stress.

The maximum stress value obtained by this method is identical to that calculated by the rules of Regulatory Guide 1.92 on page 7.3.

QUESTION:

The response addressed the effects of a dropped fuel assembly. The intent of the question was to request information on the effect of the impact of the installed fuel assemblies with the rack during a seismic event. Discuss how this effect was analyzed and the magnitude of the loads considered. This impact may not only have a local effect, but may also have a cumulative effect on the overall rack design.

ANSWER:

The most realistic model of overall fuel/box interaction is the consideration of fuel mass attached to the box structure.

The basis for this assumption is as follows:

a. Hydrodynamic coupling between fuel and box is very close. The hydrodynamic mass of the fuel was calculated to be on the order of 20 times the mass of the water displaced by the fuel.

b. Because the parameters which govern non-linear aspects of the fuel/box interaction (such as clearances, straightness, local flexibilities) are distributed statistically throughout the rack assembly, behaviour of all fuel assemblies in concert is not sufficiently probable to warrant its inclusion in a model for analysis of the design condition.

A non-linear analytical model was constructed, however, for two purposes. The first, to prove the validity of the assumption in (a.) above, i.e. that the hydrodynamically coupled model behaved similarly to the model which assumed firm attachment of fuel to box. The second, to provide an upper bound estimate of impact loadings that would result if all fuel did behave as a single mass, and that the box assembly could withstand such loadings without failure.

This model was run using the IBM 360-67 digital computer at Carnegie Mellon University. The resulting loads proved the validity of the design assumptions.

Peak base shear calculated by $k_1 (x_1 - x_3)$ max. is about 38,750 lbs. which occurs immediately following fuel impact. This load may be represented as slightly less than 1.29 g lateral, or approximately 4.14 times the design loadings. This is an upper bound estimate of impact loading if all fuel behaved as a single mass. The limiting stress intensity within the rack assembly occurs at the vee and fusion welds near the rack base and magnitudes for these stress intensities are: (for SSE conditions)

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976

$$S = 2\tau \text{ max.}$$

$$S_{vee} = 6224 \text{ psi}$$

$$S_{fusion} = 1098 \text{ psi}$$

By increasing these loads 4.14 times,

$$S_{vee\text{-}impact} = 25{,}766 \text{ psi}$$

$$S_{fusion\text{-}impact} = 4548 \text{ psi}$$

Allowable weld stress intensity limits are exceeded in the vee welds, but average fusion weld stress intensity does not exceed design allowable (S_{max} = 17,100 psi). The minimum ultimate intensity is 37,500 psi. Note that the stress intensity of the vee welds is less than this ultimate value. It is, therefore, concluded that even in the unlikely event that all fuel acts as a single mass, cumulative fuel impact loads will result in local yielding without loss of structural integrity.

QUESTION:

The response to Question 3 implies that normal code stress limits were used as acceptance criteria for all postualted load conditions. Clarify if this indeed is the acceptance criteria for your design.

ANSWER:

It was our intention that the normal code stress limits (expressed as percentages of yield stress) be used as acceptance criteria all throughout the stress analysis of the rack for all postulated load conditions.

The fuel racks and supporting structures were designed for the extreme environmental conditions occurring simultaneously with abnormal conditions, i.e., fully-loaded spent fuel racks (just after a full-core discharge) immersed in 200°F water undergoing a safe shutdown earthquake. The material yield stress, modulus of elasticity, and weld allowable stress were taken from the references at 200°F, not room temperature. By derating the allowable values in this manner, an intentional safety factor was introduced into all calculations, regardless of actual service temperature up to 200°F.

QUESTION:

The response to Question 3 references Section VIII of the ASME B & PV Code, Table II-1 references Section IX, the AISC Specification and AWS D1-1. Clarify where each of the codes is used in the design.

ANSWER:

The design of the new spent fuel racks was performed to the following procedure:

The 1974 ASME Boiler and Pressure Code, Section VIII, Division 1, was consulted for good practice concerning the welds used to fasten the fuel and water boxes together. Part UW covers the design of welded pressure vessels; specifically paragraph UW-15 details the method of weld design and calculation. Since this section was the most conservative with respect to the rest of UW, it was selected as the design criteria for all 3/32" welds having to do with the .090" box walls. The TIG arc fusion spot weld was considered as a fillet weld within a hole, and its design would be included in this section. The design allowable values were checked against tensile and bend tests of the actual weld specimens and found to be conservative by a factor of safety of six or greater. All welders and weld procedures were qualified by the applicable parts of ASME Section IX of the B & PV Code. The rest of the rack structure was treated as pressure vessel supports, the design of which was governed in Section VIII, paragraphs UG-54 and UG-22. These paragraphs referred to non-mandatory appendices D and G and to the conformance of the design to good structural practice. Appendix G suggested the AISC specifications but did not require the use of them. Since Section VIII was vague concerning the exact allowables for structural design, the AISC specifications were used since every part of the rack structure (with the exception of the 3/32" fillet weld) was considered as a structural part. All other welds were designed to the AISC Manual, Part 5 "Specifications and Codes," Section 1.17 Welds. The minimum weld size covered by the AISC was a 1/8" fillet.

The weld allowable values were taken from the appropriate table in Section VIII for the 3/32" welds. The AISC structural allowables are based on the yield stress of the material which is Type 304 stainless steel. Both formulas and tabled values are presented in AISC, Part 5 Appendix A; however, the yield stress of the rack material properties) was the government publication TID-26666 Nuclear Systems Materials Handbook, by Hanford Engineering Development Laboratory, Hanford, Washington.

QUESTION:

The response to Question 8 does not clarify the reference to floor response spectra in Item (b) at the bottom of page 4.0-3.

ANSWER:

The floor response spectra were not available from the original seismic analysis, so an alternate method of determining a conservative g-factor for analyzing the new spent fuel racks for seismic excitation was developed. A dynamic model was developed for four modules bolted together at the top for vibration in the northerly direction. This rack model was then coupled to a beam model of the reactor building based on cross-section properties and natural frequencies from the original seismic analysis. The acceleration at the center-of-gravity of the rack was determined by a model analysis using the ANSYS[1] computer program and the earthquake acceleration response spectrum for the damping values recommended in the Oyster Creek FDSAR[2]. The new spent fuel racks for Oyster Creek are assembled from 28 and 32 cell modules. These modules are assembled in various configurations by bolting them together at the top of the module. A 4x2 rack for 224 assemblies is shown schematically in Figure 1-6. The mode of vibration with the lowest frequency is in the N-S direction with the two halves in phase. Therefore, only four modules connected together at the top need be modelled to account for the lowest frequency mode of vibration which is in the N-S direction.

Each 28 cell module is modelled as a uniform cantilever beam. The module density includes the weight of the stainless steel in the rack, the fuel assemblies, and the entrained water. The material properties are those for 304 stainless steel at 200°F.

The 2-D elastic beam element, STIF3, of ANSYS accounts for both shear and bending deflections of the module. The height of the module was divided into four even sections, and the mass of each section was lumped at each end. Hence, each module has five mass points, or twenty mass points for the four modules for a total weight of 117,580 lbs

To form a 4 x 1 rack, the modules were tied together at the top by rigid beams. The frequency of the first mode of a 4 x 1 rack was 14.0 Hz. This value was considered unacceptable. See addendum to Question 9 for the derivation and frequency of a 12 x 2 rack structure.

[1]ANSYS - Engineering Analysis System, Users Manual, Swanson Analysis Systems, Inc., Elizabeth, Pa.

[2]Oyster Creek Nuclear power Plant Unit No. 1 - Facility Description and Safety Analysis Report". Figure V-3-1 and Table V-3-1.

*Addendum 1 to Supplement 1 to Amendment 76 November, 1976.

A cantilever beam model of the reactor building was developed from cross-sectional properties given by the original seismic analysis. This analysis tabulated the cross-sectional areas, moments of inertia and weights for various elevations. A Young's Modulus of 1.2×10^6 lb/in^2 was used to match the first three natural frequencies of the original analysis. A comparison of the natural frequencies is shown below:

	f_1	f_2	f_3
Original Model - 1965	3.71 Hz	6.17 Hz	18.7 Hz
JCP&L Model - 1976	3.73 Hz	6.24 Hz	17.4 Hz

To perform a response spectrum analysis, the 4 x 1 fuel rack model was coupled to the reactor building model (see Figure 8-1). Nodes 1, 11, 21, 31 and 95 all have the same displacements, i.e., the bottom of the spent fuel pool and the bottom of the fuel racks experience the same vibration.

Figure V-3-1 of the Oyster Creek FDSAR is the earthquake acceleration response spectrum for 0.11g ground motion (OBE) for various values of damping. The accelerations were doubled to obtain response spectra for 0.22g ground motion (SSE).

Table V-3-1 of Section 3, Structural and Shielding Design of the Oyster Creek FDSAR, gives the recommended damping factors for various types of structures: 2% for bolted and riveted structures, and 10% for reinforced concrete. Values for 2% were used in the neighborhood of the rack resonance, i.e., 10-16 Hz, and values for 10% were used for the reactor building resonances.

The acceleration response for the center-of-gravity of the racks was obtained by adding the model responses for the first ten modes by the square root of the sum of the squares (SRSS) method. This result was combined with the acceleration for the rocking mode from the original seismic analysis (0.1318g) by the SRSS to give a total acceleration at the C.G. of the rack as 0.306g.

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976.

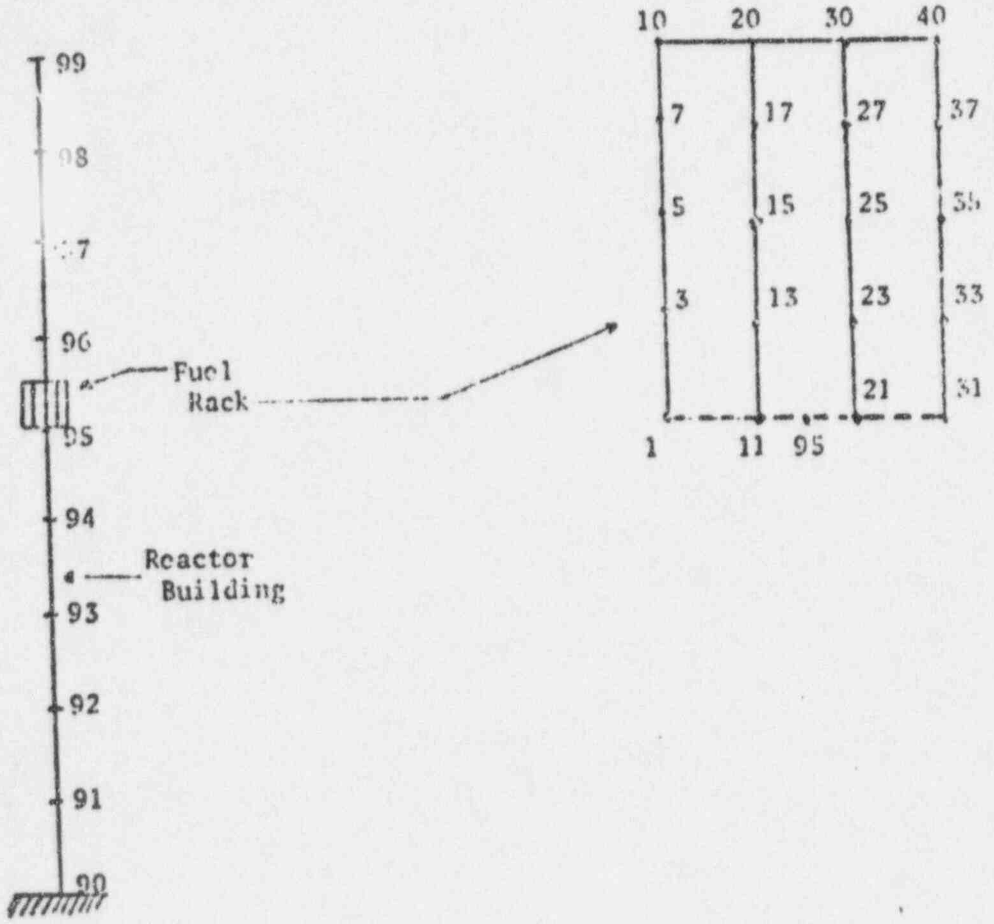


FIG. 8-1 COMPUTER MODEL OF REACTOR BUILDING AND FUEL RACK WITH NODE POINTS

QUESTION:

Clarify if the frequencies calculated in response to Question 9 included the added mass due to submergence in water. Quantify the mass which was used. Furthermore, clarify what a "4 x 2 rack structure" is and why it was chosen for analysis.

ANSWER:

The frequency analysis 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. The average percentages of the constituents of the total rack structure mass were: 13 percent steel deadweight, 62 percent fuel assembly deadweight, and 25 percent for all trapped water and submergence effects.

During the preliminary design calculations, it was discovered that a single 28 cell rack, fastened to its base and including all fuel assemblies and water effects, had a first mode vibration frequency of approximately eight Hertz. This frequency was judged as being unacceptably low. Further analysis showed that the first mode frequency of vibration could be increased by mechanically interconnecting several racks together. Sufficient cases were analyzed until a "4 x 2 rack structure" configuration provided an acceptable first mode frequency. This was based on racks being tied together at two elevations.

Due to potential installation problems it was later decided that the racks be tied together at the top only. Subsequently, a "12 x 2 rack structure" was adopted to obtain a more acceptable first mode frequency. Its configuration is shown in Figure 9-1.

The results of analysis show that a "12 x 2 rack structure" has a frequency of 43.9 Hz for the N-S direction.

The substitution of a 32 fuel assembly (32 FA) rack for any or every 28 FA rack does not significantly change the first mode frequency of the "12 x 2 structure". This configuration is designed to stand along without any other lateral supports and meet the seismic criteria. To insure acceptable rack frequency, a minimum 24 (12x2) racks will be joined together in the Oyster Creek spent fuel pool.

*Addendum 1 to Supplement 1 to Amendment 78 November, 1976.

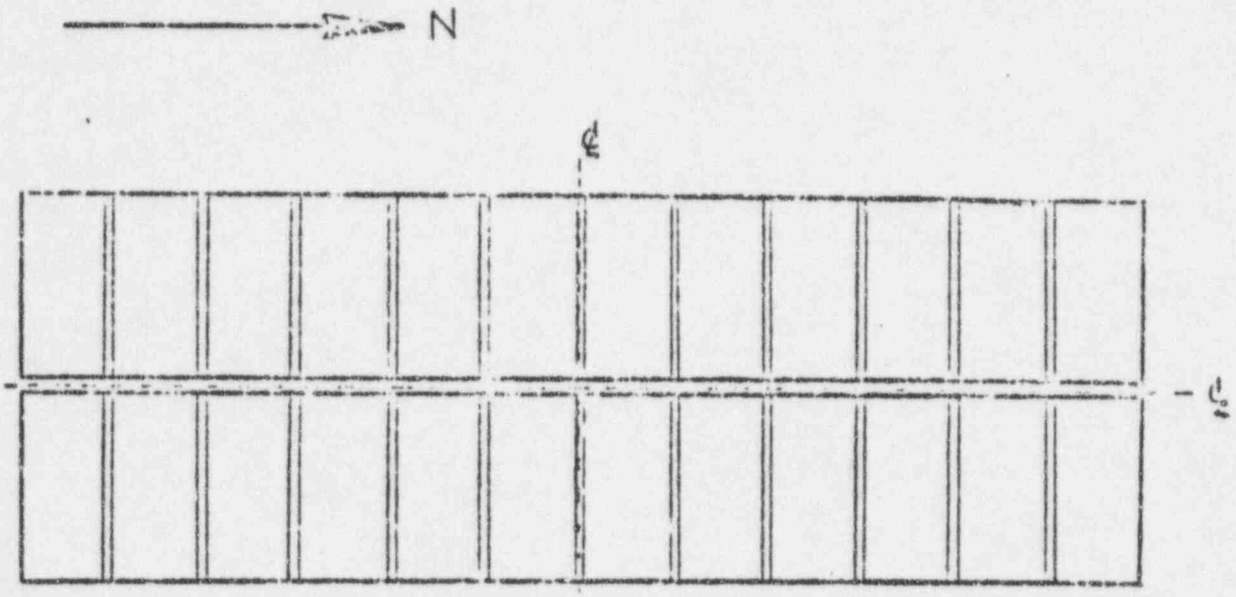


FIG. 9-1 - 12 x 2 RACK STRUCTURE

Question:

Propose Technical Specifications to limit the cask vertical movement above the top plate of the Cask Drop Protection System to 6 inches or less. Also propose Technical Specifications to limit the weight of loads moved over stored fuel in the spent fuel pool.

Answer:

The vertical movement of the cask on the refueling floor is limited to ≤6 inches above the refueling floor by procedural control. It is limited to ≤6 inches above the top plate of the cask drop protection system by vertical limit switches. Rail stops limit cask movement in the North, South, and West direction. These limitations are discussed in Supplement No. 1 to Amendment 68 of the FDSAR.

A proposed Technical Specification change to limit the cask vertical movement to ≤6 inches above the top plate of Cask Drop Protection System is submitted with this Addendum No. 1 to Supplement No. 1 to Amendment No. 78.

Also included in this Addendum is a proposed Technical Specification Change to limit the maximum weight of loads moved over the stored fuel in the spent fuel pool of a fuel assembly.

Question:

When will the temporarily stored waste materials in the spent fuel pool be removed?

Answer:

A waste removal project is currently underway to remove materials such as the channel clips, LPRM's, etc. It is anticipated that these materials will be removed from the spent fuel pool prior to rack installation. This cleanup will be continued as the pool areas are cleaned of crud, if needed, before rack installation.

Question:

Provide the best estimate of the exposure due to spent fuel pool activities.

Answer:

Estimates of the exposure for typical work around the spent fuel pool area on the refueling floor are 8 man-days/week with 30 mrem/day/man for a total of ~ 12.5 rem/year. This typical exposure is based on past experiences and includes routine operations but does not include spent fuel rack replacement. As noted before, the pool area will be cleaned before rack installation, and crud build-up will be cleaned in this operation.

If divers are needed for the installation of the racks, a maximum dose of 15 man-rem is expected for the divers during the installation. No diver will exceed the requirements of 10CFR20 during the operation.

Question 47:

How much stainless steel will be used in the rack fabrication?

Answer:

Approximately 300,000 lb of stainless steel will be used in the construction of 61 racks which will have storage capacity for 1800 fuel assemblies.

Exhibit 8

U.S. DEPARTMENT OF COMMERCE
National Technical Information Service
NUREG-0123-REV-2

Standard Technical Specifications for General Electric Boiling Water Reactors, Revision of August 1979

(U.S.) Nuclear Regulatory Commission, Washington, DC

Sep 79

NUREG- 0123-REV.-2

NUREG-0123
Revision 2

STANDARD TECHNICAL SPECIFICATIONS FOR GENERAL ELECTRIC BOILING WATER REACTORS

Revision of August 1979

Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

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Standard Technical Specifications for General Electric Boiling Water Reactors

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3. RECIPIENT'S ACCESSION NO.
N/A

7. AUTHOR(S)
Robert R. Bottimore

5. DATE REPORT COMPLETED
MONTH August | YEAR 1979

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Operating Reactors
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U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

DATE REPORT ISSUED
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12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO
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11. CONTRACT NO.
N/A

13. TYPE OF REPORT
Technical Report

PERIOD COVERED (Inclusive dates)
N/A

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

The Standard Technical Specifications for General Electric Boiling Water Reactors (GE-STS) is a generic document prepared by the USNRC for use in the licensing process of current General Electric Boiling Water Reactors. The GE-STS sets forth the Limit, Operating Conditions and other requirements applicable to nuclear reactor facility operation as set forth by Section 50.36 of 10 CFR 50 for the protection of the health and safety of the public. This document is revised periodically to reflect current licensing requirements.

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NRC FORM 335 (7-77)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of (2500) pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of (2500) pounds over fuel assemblies in the spent fuel storage pool racks shall be demonstrated OPERABLE within 7 days prior to and at least once per 7 days during crane operation.

Exhibit 9

Consumers Power

POWERING MICHIGAN'S PROGRESS

Palisades Nuclear Plant: 27780 Blue Star Memorial Highway, Covert, MI 49043

May 13, 1996

U S Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
RESPONSE TO NRC BULLETIN 96-02: MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY RELATED EQUIPMENT

On April 11, 1996, the NRC issued Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipmeht." This bulletin requested the licensee to perform a review of their plans and capabilities to handle heavy loads while the reactor is at power in accordance with existing regulatory requirements. It further requests that if any planned activities are determined to be outside the plant licensing basis, then changes to the operating license and/or technical specifications are to be identified and subsequently processed for reviews and approvals. This submittal provides the requested 30 day response to NRC Bulletin 96-02.

The control of heavy load movements at Palisades, when the reactor is at power, was reviewed to determine the compliance to licensing basis requirements. Based on that review, we have determined that no previous occurrence or future plan to handle heavy loads is inconsistent with the established licensing basis. The attachment to this letter contains specific responses to the NRC Bulletin 96-02 requested information.

JE11 1/1

SUMMARY OF COMMITMENTS

This letter contains no new commitments and no revisions to existing commitments.

Richard W. Smedley
Manager, Licensing

CC Administrator, Region III, USNRC
Project Manager, NRR, USNRC
NRC Resident Inspector - Palisades

Attachment

CONSUMERS POWER COMPANY

To the best of my knowledge, the contents of this response to NRC Bulletin 96-02, which required that Palisades review their plans and capabilities to move heavy loads over spent fuel, over fuel in the reactor core, or over safety related equipment when the reactor is at power, are truthful and complete.

By ______________________
Richard W. Smedley
Manager, Licensing

Sworn and subscribed to before me this 13th day of May 1996.

____
Alora M. Davis, Notary Public
Berrien County, Michigan
(Acting in Van Buren County, Michigan)
My commission expires August 26, 1999

[SEAL]

ATTACHMENT

CONSUMERS POWER COMPANY
PALISADES PLANT
DOCKET 50-255

CONSUMERS POWER COMPANY'S RESPONSE TO NRC BULLETIN 96-02

MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT

NRC BULLETIN 96-02
MOVEMENT OF HEAVY LOADS OVER SPENT FUEL, OVER FUEL IN THE REACTOR CORE, OR OVER SAFETY-RELATED EQUIPMENT

NRC Requested Action:

Pursuant to Section 182a, the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), all addresses must submit the following written information:

(1) For licensees planning to implement activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment within the next two years from the date of this bulletin, provide the following:

> *A report, within 30 days of the date of this bulletin, that addresses the licensee's review if its plans and capabilities to handle heavy loads while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) in accordance with existing guidelines. The report should also indicate whether the activities are within the licensing basis and should include, if necessary, a schedule for submission of a license amendment request. Additionally, the report should indicate whether changes to Technical Specifications will be required.*

Consumers Power Company Response:

The control and handling of heavy loads at Palisades, with the reactor at power, was reviewed to determine compliance with licensing basis requirements. Based on that review, we have determined that no previous occurrence or future plan to handle heavy loads would be inconsistent with established licensing basis.

The movement of heavy loads in the containment building area is restricted by Technical Specification 3.21.1 to periods when the Primary Coolant System temperature is below 225°F. There are no plans to change this restriction. However, there is a submittal under NRC review at this time that would remove the Heavy Loads Requirements from Palisades Technical Specifications and transfer the requirements to a Palisades administrative document. Heavy Loads requirements do not meet the criteria of 10 CFR 50.36 and are not addressed in Standard Technical Specifications, NUREG 1432.

The movement of heavy loads in the Spent Fuel Pool Area is restricted by Technical Specification 3.21.2, which includes the following requirements:

a. Heavy loads shall not be moved over fuel stored in the main pool zone.

b. Heavy loads shall not be moved over areas of the main pool zone which do not contain fuel unless the fuel stored in the main pool zone has decayed a minimum of 30 days when the charcoal filter is operating; or, the fuel stored in the main pool zone has decayed a minimum of 90 days when the charcoal filter is not operating.

c. Heavy loads shall not be moved over the north tilt pit zone unless the fuel stored in the north tilt pit zone has decayed a minimum of 22 days when the charcoal filter is operating; or, the fuel in the north tilt pit zone has decayed a minimum of 77 days when the charcoal filter is not operating.

d. Heavy Loads shall not be moved over the 649' level of the auxiliary building unless:

   1. The fuel storage building crane interlocks are operable or they are bypassed and the crane is in administrative control of a supervisor, and

   2. No fuel handling operations are in progress.

e. Loads weighing more than 25 tons shall not be moved over the main pool zone unless an evaluation in compliance with section 5.1 of NUREG-0612 has been completed.

f. Heavy loads shall not be moved unless the potential for a load drop is extremely small, as defined by Generic Letter 85-11, or an evaluation in compliance with section 5.1 of NUREG-0612 has been completed.

g. The Fuel Pool Building Crane shall not be used to move material past the fuel storage pool when its interlocks are inoperable.

The following heavy loads are moved in accordance with the existing licensing requirements over the main fuel pool zone that does not contain fuel:

a. Dry Fuel Storage (DFS) Transfer Cask
b. DFS Transfer Cask Lifting Rig
c. DFS Multi-Assembly Sealed Basket (MSB)
d. MSB Shield Cover
e. MSB Structural Cover
f. Ultrasonic Fuel Inspection Rig

During the loading of dry fuel storage casks at Palisades, there are three heavy loads, the DFS Shield Cover, Structural Cover and MSB/Transfer Cask Lifting Rig, that must

be moved over the fuel that has been placed into the DFS MSB/Transfer Cask during transport activities. Based on several Palisades-NRC conversations in November of 1992, the fuel in the MSB was determined to not be subject to the requirements of Technical Specification 3.21.2. The rationale for this interpretation was agreed to at that time by the Palisades NRC Project Manager, NRC Legal Department, and the Senior Resident Inspector. The rationale included the following factors:

a. Technical Specification 3.21.2.a is applicable only to fuel stored in the main pool zone. Fuel in the MSB is not being stored in the main pool zone but is in transit.

b. As required by the Palisades Technical Specifications, fuel stored in the main pool zone is required to be stored in either Region I or Region II racks under conditions (minimum burn up and boron concentration levels) which do not apply to fuel in the MSB.

c. Fuel placed in the MSB is subject to the requirements of Attachment A to the Certificate of Compliance.

NRC BULLETIN 96-02

(2) *For licensees planning to perform activities involving the handling of heavy loads over the spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) and that involve a potential load drop accident that has not previously been evaluated in the FSAR, submit a license amendment request in advance (6-9 months) of the planned movement of the loads so as to afford the staff sufficient time to perform an appropriate review.*

Consumers Power Company Response:

No heavy load movements over spent fuel, fuel in the reactor core, or safety related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) are planned that have not been previously evaluated. The process of placing spent fuel into dry storage casks is described in the FSAR. The heavy load activities associated with the cask loading process are not specifically addressed in the FSAR; however, the heavy load activities have been evaluated as part of the spent fuel loading process.

NRC BULLETIN 96-02

(3) *For licensees planning to move dry storage casks over spent fuel, fuel in the reactor core, or safety related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled) include in item 2 above, a statement of the capability of performing the actions necessary for safe shutdown in the presence of radiological source term that may result from a breach of the dry storage cask, damage to the fuel, and damage to safety related equipment as a result of a load drop inside the facility.*

Consumers Power Company Response:

Reference the reply to item 2 above.

NRC BULLETIN 96-02

(4) *For licensees planning to perform activities involving the handling of heavy loads over spent fuel, fuel in the reactor core, or safety-related equipment while the reactor is at power (in all modes other than cold shutdown, refueling, and defueled), determine whether changes to Technical Specifications will be required in order to allow the handling of heavy loads (e.g., the dry storage canister shield plug) over fuel assemblies in the spent fuel pool and submit the appropriate information in advance (6-9 months) of the planned movement of the loads for NRC review and approval.*

Consumers Power Company Response:

There are no plans to handle heavy loads over spent fuel, fuel in the reactor core, or safety related equipment while the reactor is at power which require changes to Technical Specifications. Refer to the reply to items 1 and 2 for identification of heavy loads that are moved over the main fuel pool zond areas which do not contain spent fuel and the three heavy loads that are moved over fuel that is in transit and contained in the MSB/Transfer Cask.

DOCKETED
USNRC
November 15, 1996

'96 NOV 19 P3:30

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-219-OLA
GPU NUCLEAR CORPORATION	)	(Tech. Spec. 5.3.1.B)
	)	
(Oyster Creek Nuclear Generating Station)	)	ASLBP No. 96-717-02-OLA

CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Motion for Summary Disposition," "Statement of Material Facts as to Which There is no Genuine Dispute," and "Affidavit of John C. Fornicola" dated November 15, 1996, were served upon the persons listed below by deposit in the United States mail, first class, postage prepaid, or, where indicated by an asterisk, by hand delivery, this 15th day of November, 1996. In addition, where marked by two asterisks, a copy of the pleadings (without exhibits) was transmitted by facsimile or e-mail.

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