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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	Docket Nos. 50-254-SP
COMMONWEALTH EDISON COMPANY	)	50-265-SP
(Quad Cities Station,	)	(Spent Fuel Pool Modification)
Units 1 and 2)	)	

Dear Administrative Judges:

Enclosed is Supplement 6 to the Licensing Report prepared by Joseph Oat Corporation for Commonwealth Edison Company.

Very truly yours,

Robert G. Fitzgibbons Jr.

RGF:emc
Enclosure
cc Service List

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December 8, 1981

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Quad Cities Station Units 1 and 2
Transmittal of Supplemental 6 to
Revision 1 of the Licensing Report
on High Density Fuel Racks
NRC Docket Nos. 50-254/265

Reference (a): T. A. Ippolito letter to J. S. Abel dated May 15, 1981.

(b): T. A. Ippolito letter to J. S. Abel dated May 18, 1981.

Dear Mr. Denton:

Enclosed is Supplement 6 to Revision 1 of the report prepared by Joseph Oat Corporation for Commonwealth Edison entitled "Licensing Report on High Density Spent Fuel Racks for Quad Cities Units 1 and 2." This supplement provides responses to a portion of the questions provided in References (a) and (b). These responses are for questions numbered as follows:

12.1.1 (Ref. a)
12.1.6 (Ref. a)
12.1.7 (Ref. a)
12.1.8 (Ref. a)
12.2.3c (Ref. b)
12.2.3d (Ref. b)
12.2.3g (Ref. b)
12.2.3i (Ref. b)

In addition, Supplement 6 provides Section 7 concerning "Accident Analysis".

Please address any questions you may have concerning this matter to this office.

Dupe B
~~8112100349~~

One (1) signed original and thirty-nine (39) copies of this transmittal are provided for your use.

Very truly yours,

Thomas J. Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

Enclosure

cc: Region III Inspector - Quad Cities

TJR/lm

2787N

7. ACCIDENT ANALYSIS

7.1 Introduction

The Quad Cities Station Safety Analysis Report, and other documents, have presented results of analyses of several types of accidents which could potentially affect the spent fuel storage pools. Installation of the proposed high density racks will enable Licensees to store increased amounts of spent fuel in the Quad Cities spent fuel pools. Accordingly, accidents involving the spent fuel pools have been reevaluated to ensure that the proposed spent fuel pool modification does not change the present degree of assurance of public health and safety. The following accidents have been considered:

- Fuel Pool - Earthquake Loading
  Loss of water
- Cask Drop
- Reactor Building - Earthquake Loading
  Tornado Loading & Missiles 6
- Chimney - Wind Loading
- Refueling Accidents - Dropped Fuel
  Dropped Gates
  Dropped Channel Measuring Device
- Radwaste Leaks and Spills
- Turbine Missiles

7.2 Results of accident reevaluation

7.2.1 Fuel Pool

The effects of earthquake loadings on the fuel racks and spent fuel pool floor are discussed in Sections 6.0 and 9.0 respectively of this report. The loss of cooling water in the spent fuel pool is discussed in report Section 5.1.2.

7.2.2 Cask Drop

On May 31, 1973, Commonwealth Edison submitted Dresden Station Special Report No. 28 to the NRC. The report stated that it was applicable for Quad Cities Station also. Addendums 1 and 2 were submitted on July 2 and August 10, 1973. This report, "Analysis and Procedures for Handling General Electric 1F-300 Spent Fuel Shipping Cask", contained "Cask Drop Analyses". The report was accepted by the NRC letter of September 10, 1973, from D. J. Skouholt to J. S. Abel. The report concluded that the fuel pools could withstand a drop of the 1F-300 cask.

Subsequent to Special Report No. 28, modifications were made to the Reactor Building crane which preclude postulated drops of a 100-ton-spent fuel shipping cask. These modifications are described in Quad-Cities Special Report No. 16 transmitted by letter from Commonwealth Edison Company to the NRC dated November 8, 1974. Supplementary information was transmitted to the NRC by
letters dated June 10 and December 8, 1975 and February 9, March 6
2, and March 29, 1976. The NRC approved the modifications and associated changes in the Technical Specifications in the letter of January 27, 1977 to Commonwealth Edison Company. Therefore, the reracked spent fuel pool will not be subject to a cask drop accident analysis.

7.2.3 Reactor Building

The ability of the reactor building to resist earthquakes and tornadoes has not been affected by the spent fuel pool reracking, except as described in Section 9.0 of this report. Therefore, except for the described differences, the information presently contained in the FSAR is still valid.

7.2.4 Chimney

The accident involving the chimney is described in Quad Cities FSAR Section 12.2.1.2. The scenario described therein is that the top 250 feet of the chimney break off during a tornado and fall towards the reactor building. The end 3 feet of the stack smash into the Unit 2 reactor shield plug, breaks up, and pieces fall into the spent fuel pool. Some of the concrete pieces would be stopped from going into the pool by the raised pool curbing and guard rails. Some could go over or through these barriers. The size and weight and velocity of the pieces which could land on the top of the fuel racks would be less than that of other objects (such as fuel assemblies, canal gates, and a channel measuring device) which have been analyzed. Therefore, the chimney accident is not considered to be a limiting one or a safety hazard.

7.2.5 Refueling Accidents

This section considers three (3) accidents associated with the handling of fuel assemblies, the movement of transfer and reactor canal gates and the use of a fuel channel measuring device. No other objects are ever moved over spent fuel. 6

7.2.5.1 The consequences of dropping a new or spent fuel assembly as it is being moved over stored fuel is discussed in Sections 6.6 and 4.4.6 of this report. These analyses concluded that the postulated accidents will not result in a K_{eff} above that calculated for the rack design.

7.2.5.2 The reactor canal to pool gate is conservatively assumed to fall from an elevation of 2' above the rack module. The gate is constructed of aluminum, and weights 900 lbs. in air. It is the largest and heaviest gate. Its minimum frontal areas corresponds to an upright vertical fall.

The mathematical model constructed to determine the impact velocity of the above falling object is based on several conservative assumptions, such as

a. The virtual mass of the body is conservatively assumed to be equal to its displaced fluid mass. Evidence in the literature[1] indicates that the virtual mass can be many times higher.

b. The minimum frontal area is used for evaluating drag coefficient.

c. The drag coefficient utilized in the analysis are lower bound values reported in the literature.[2] In particular, at the beginning of the fall when the velocity of the body is small, the corresponding Reynolds number is low resulting in a large drag coefficient.

d. The falling bodies are assumed to be <u>rigid</u> for the purposes of impact stress calculation on the rack.

The solution of the body motion problem is found analytically. The impact velocity thus computed is used to determine the maximum stress generated due to stress wave propagation.

The gate drop accident creates a local stress on the rack cell edge of 17,000 psi. Since this value is lower than the material yield point, no deformation will occur, and no change in K_{eff} will occur. 6

7.2.5.3 The channel measuring device weights 1000 lbs. This device is made of aluminum except for some small 304 stainless steel components. The main components in this device are an 8" x 30' - 6" long aluminum I-beam and a gasketed 12" x 27" shaped base plate. This device is conservatively assumed to free fall from an elevation of 30' above the racks, and impact the top edge of one panel of a rack module. Since the minimum frontal surface of the device corresponds to an upright position, we assume conservatively that this object remains vertical during its fall. If it does not remain vertical, the effects would be lessened.

The results of the analysis show that for the channel measuring device, the maximum calculated stress is 36,500 psi. Since the yield strength of the rack edge material is 25,000 psi, there will be plastic deformation of the rack of about 1/8 inch. This amount of deformation will not impair the rack's ability to maintain a K_{eff} equal or less than 0.95.

7.2.6 Radwaste Leaks and Spills

It has been determined that the spent fuel pool modification will not result in increase usage of the pool cleanup system (see Section 8.0). Therefore, the analyses presented in the FSAR are still valid.

7.2.7 Turbine Missiles

This postulated accident has been determined to not be a credible one for a BWR such as Quad Cities and the reracking does not affect this conclusion. 6

<u>References to Section 7</u>

1. "Standards of Tubular Exchanger Manufacturers Association", 6th edition, Section 12.

2. "Fluid Mechanics" by M. C. Potter and J. F. Foss, Ronald (1975), p. 454.

12. RESPONSES TO NRC QUESTIONS

Given below are NRC questions concerning the Licensing Report on High-Density Spent Fuel Racks for Quad Cities Units 1 and 2. They are listed by date of transmittal. Also given below are responses to those questions or the word "Later" indicating that the response will be communicated at a later date.

12.1 Questions from T. A. Ippolito to J. S. Abel transmitted on May 15, 1981

12.1.1 Question:

As a result of replacing the fuel pool racks, there is an appreciable increase in the applied load to the fuel pool concrete floor. Indicate the method and the code used in the analysis of the concrete fuel pool slab.

Response: The method and codes used in the analysis of the concrete fuel pool slab are contained in Supplement 5 to Revision 1 of Licensing Report Section 9.0, Pool Structural Calculations, submitted to the NRC on November 2, 1981. 6

12.1.2 Question:

Provide the floor response spectra or the time history used in the analysis of the spent fuel racks and state the source of this information.

Response: Section 6.7 of Supplement 4 to Rev. 1 of the Licensing Report submitted on 10/19/81 gives the source of the time history data. Figures 6.9 and 6.10 of Section 6.7 depict horizontal and vertical pool floor accelerations used in the racks analyses.

12.1.3 Question:

Indicate the damping value used in the analysis of spent fuel racks and state whether this value conforms with Regulatory Guide 1.61.

Response: Paragraph 6.2.4 of Supplement 4 to Rev. 1 of the Licensing Report submitted on 10/19/81 states that 1% damping was used in the analysis of the spent fuel racks. This value is consistent with that used in the FSAR and conservative with that permitted by Regulatory Guide 1.61.

12.1.4 Question:

Indicate whether material, fabrication, installation, and quality control conform with the ASME code, Subsection NF.

Response: Yes, material, fabrication, inspection and quality control conforms with ASME code, Subsection NF.

12.1.5 Question:

Indicate if there is any possibility that the shipping cask may drop onto the fuel pool liner or on to the fuel pool racks and what design considerations are given to the fuel pool liner and racks.

Response: Section 10.1.2 of the Quad-Cities FSAR describes the fuel pool structure and leak detection system. In regard to cask drop this section references the Dresden-2/3 FSAR (Dockets 50-237/50-249) Amendment 16/17, Section 11, Fuel Pool Damage Protection. In response to NRC question 2.9.3.11, Section 10 of Amendment 11 of the Quad-Cities FSAR describes the fuel pool liner design and additional details of the leakage detection system. Dresden Special Report No. 28 transmitted to the NRC from Commonwealth Edison by letter dated May 31, 1973, provides a structural analysis which concludes that a dropped cask will not penetrate the bottom of the pool. This report also applies to Quad-Cities. Addenda Nos. 1 & 2 transmitted to the NRC by letters dated July 2, 1973 and August 10, 1973 provide additional information.

Modifications have been made to the Reactor Building crane handling system which preclude postulated drops of a 100-ton-spent fuel shipping cask. These modifications are described in Quad-Cities Special Report No. 16 transmitted by letter from Commonwealth Edison Company to the NRC dated November 8, 1974. Supplementary information was transmitted to the NRC by letters dated June 10 and December 8, 1975 and February 9, March 2, and March 29, 1976. The NRC approved the modifications and associated changes in the Technical Specifications in the letter of January 27, 1977 to Commonwealth Edison Company.

12.1.6 Question:

Provide the names of the codes and standards used in the fuel pool liner design.

Response: The liner was designeed in 1968 to ASME Section VIII, Subsection B, Part UW and ASME Section IX. Weld Procedure Qualifications were made in accordance with Section VIII, Q10 thru Q19.

All exposed plate, shapes, and hardware was purchased in 1968 to 6
ASTM A167, Type 304. The floor is ¼" plate while the walls are 3/16" plate. All plate was purchased hot rolled, annealed, and pickled followed by cold rolling and polishing.

12.1.7 Question:

With regard to the fuel assembly drop on the top of the rack, provide the following:

a. Detailed description of the method used to satisfy the acceptance criteria for dropped fuel accident I and II.

b. Comparison between drops in the tilted position, straight drop and on the corner of the rack.

c. Indicate whether other modes of failure of the racks exist beside crushing.

Response: a. The method used in analyzing the type I and II dropped fuel accidents was the classical one of a rigid body impacting the end of a plate or solid rod. The fuel assembly was assumed to be rigid and the drag effects of the water mass were conservatively ignored. Thus, the impact velocity could be determined by elementary dynamics. The impact of the rigid body fuel assembly on the fuel cell lead-in edge and base plate was used to determine the maximum magnitude of stress induced in these members.

b. A straight drop hitting the top of the rack or the base plate of the rack produces the highest local stress levels in the rack. The results of the fuel assembly drop analyses were previously given in the response to NRC question 12.2.3.h. 6

c. Crushing and wide spread plastic deformation were the modes of failure examined. Plastic deformation could cause the geometric dimensions and tolerances used in the criticality analysis to be violated, but did not occur for the cases analyzed.

12.1.8 Question:

Indicate in detail the methodology used to demonstrate the leaktight integrity of the fuel pool liner when subjected to either the postulated fuel assembly drop or the cask drop over the spent fuel pool liner. The heavy drop should be analyzed for the tilted position and straight drop.

Response: The methodology and results of a fuel assembly drop within a fuel storage cell are described in Section 6.6 of the Licensing Report and further clarified in the responses to NRC questions 12.1.7 and 12.2.3.h. The consequences of dropping a fuel assembly outside of a fuel storage cell are described in the FSAR. The probability of such an accident will be reduced considerably when the new high 6 density racks are installed as much more of the floor liner will be protected by the racks.

The cask drop over the spent fuel pool liner is addressed in Section 7.0 of the Licensing Report and further discussed in the response to NRC question 12.1.5. The results of a cask drop accident are not affected by the modification.

12.1.9 Question:

Indicate whether the proposed fuel storage pool modifications conform with the staff position on "Fuel Pool Storage and Handling Application", dated April, 1978, including revisions dated January, 1979. If any deviations exist, identify and justify these deviations.

Response: Yes, the guidance is followed, with the exception of the Technical Specification for maximum enrichment. This is because of the variety of enrichments in the fuel and the existance of the subcriticality specification of k_{eff} less than or equal to 0.95.

12.1.10 Question:

The seismic analysis as presented in the submittal is not clear. Indicate in detail how all the seismic models and parameters (Figure 6.1, 6.3, 6.4, 6.5, 6.6, 6.7 and 6.8, the friction forces and floor response spectra) fit together to predict the seismic stresses. Indicate the interrelationship among the models.

Response: See Revision 1 to Chapter 6, Seismic Analyses Description, submitted to the NRC by letter from T. J. Rausch to H. R. Denton on June 24, 1981.

12.1.11 Question:

Because different type modules were used in the proposed modification with different sizes and weights, indicate which type was used in the seismic and sliding analysis. Indicate also how other types were qualified for the postulated loadings.

Response: Section 6.7 of Supplement 4 to Rev. 1 of the Licensing Report submitted on 10/19/81 indicates rack types, sizes, and weights used in the seismic and sliding analyses.

12.2 Questions from T. A. Ippolito to J. S. Abel transmitted on May 18, 1981

12.2.1 Question:

When Section 5.1, Heat Generation Calculations, is provided, include the following information:

a. Indicate the minimum elapsed time between shutdown and when the dis charged fuel is in the spent pool for all anticipated fuel discharge cycles.

Repcnse: See Section 5 of Supplement 2 to Revision 1 of the Licensing Reports submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

b. For Units 1 and 2 spent fuel pools, indicate the number of fuel assemb lies and their respective decay times of all fuel that will be in the pools when reracking occurs.

Response: See Revision 1 of Licensing Report submitted to the NRC by letter from T. J. Rausch to H. R. Denton on June 24, 1981.

c. It is noted in the FSAR that portions of the RHR system may be used to augment the spent fuel pool cooling system by inserting spool pieces in the spent fuel pool cooling lines shown in Figure 10.2.1. In this regard, indicate the length of time required to install these spool pieces and describe the capability of the RHR system to remove the heat from the spent fuel pool over a range of pool temperatures and with and without the spent fuel pool cooling system in operation.

Response: The time required to install the spool pieces is discussed in the response to question 12.2.2. The capability of the RHR system to remove heat from the spent fuel pools is discussed in Section 5 of Supplement 2 to Revision 1 of the licensing report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

d. For Units 1 and 2 indicate the length, width and depth of the spent fuel pools and the minimum volume of water in each when all storage racks are filled with fuel assemblies.

Response: As shown in Section 2 of the licensing report, the length and width of each pool are 41 feet and 33 feet respectively. The depth of water in each pool is 39 feet. As stated in Section 5 of Supplement 2 to Revision 1 of the licensing report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981, the water inventories in the Quad-Cities Unit 1 and Unit 2 spent fuel pools are 44887 and 44471 cubic feet respectively when all racks are in place in the pools and every storage location is occupied.

e. Figure 2.1 and 2.2 of the March 26, 1981 submittal shows that the downcomer region, i.e., space between the racks and walls of the pool, is quite small. Further, the vertical dimension of the water plenum formed by the base plate of storage racks and the pool bottom is 6-1/2 inches. Assuming the maximum heat load is adversely located in the storage racks demonstrate that sufficient circulation will occur to preclude nucleate boiling.

Response: See Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

12.2.2 Question:

Assuming the reactor is operating at power when it becomes necessary to utilize the RHR system to cool the spent fuel pool, describe and discuss the steps that must be taken and the elapsed time before the RHR system can be placed in the fuel pool cooling mode of operation.

Response: Using the Residual Heat Removal (RHR) System for fuel pool cooling will render one of the two loops (two pumps and one heat exchanger) unavailable for use in any of the safety functions (LPCI or containment cooling). Quad Cities Technical Specifications allow LPCI and one loop of containment cooling to be inoperable during reactor operation as long as 1) the other loop of containment cooling is available, both core spray systems are operable, and both diesel generators are operable, and 2) the loop used for fuel pool cooling is returned to norma' within seven days, or the reactor shall be shut down.

Once it has been determined that supplemental fuel pool cooling using RHR is necessary, the RHR/LPCI Mode Outage Report Surveillance would be performed, and crews would be dispatched to install the two spool pieces which join the fuel pool cooling system to RHR. When this has been accomplished, the valving operations may begin. This involves the closing of several motor-operated valves, racking out the breaker on another motor-operated valve, and the opening of two manual valves near the fuel pool cooling heat exchangers. Next, the RHR Shutdown Cooling Mode suction header must be filled and vented and the RHR system vented. Finally, the RHR service water system is started and an RHR pump is started to commence fuel pool cooling. The total elapsed time would be approximately three hours if two maintenance crews were available (one for each spool piece) or four hours if a single crew installed both spool pieces. At times when no maintenance crew is on site, an additional one to two hours would be required to assemble the necessary personnel.

12.2.3 Question:

For both Units 1 and 2 spent fuel pool reracking operations, provide the following additional information:

a. Assuming a load drop, describe and discuss, with the aid of drawings, the travel paths of the new and existing storage racks with respect to plant equipment that may be needed to attain a cold safe shutdown or to mitigate the consequences of an accident.

Response: Diagrams will be prepared before moving racks based upon results of NUREG-0612 studies.

b. Provide the weights of the racks. Describe and demonstrate the ade quacy of the lifting rig attachment points, on the new and old racks, to withstand the maximum forces that will be experienced during the load handling operations.

Response: The weight of the racks is contained in the Revision 1 Licensing Report submitted to the NRC on June 24, 1981 by letter from T. J. Rausch to H. R. Denton. Lifting rig requirements are not yet defined and will be submitted later.

c. With the aid of a drawing, describe the lifting rigs that will be employed in handling the racks and demonstrate their adequacy.

Response: The lifting rigs that will be used to handle the fuel racks are described in Supplement 5 to Revision 1 of Licensing Report Section 3.3 submitted to the NRC on November 2, 1981. Figures 3-7 and 3-8 show these rigs. Both lifting devices will be analyzed to assure their adequacy to safely handle the fuel racks. 6

d. Assuming stored spent fuel is in the pool when the storage racks are being removed or installed, demonstrate that the stored spent fuel is not within the area of influence of dropped racks should one or more of legs of the lifting rig fails.

Response: An installation sequence has been developed whereby there will be no old or new racks containing stored fuel immediately adjacent to the location where a rack is being lifted. Therefore, a rack can be dropped vertically or assume a hanging angle and still not land on stored fuel. 6

e. FSAR Figures 12.1.1 and 12.1.2 shows a transfer canal joining Unit 1 pool with Unit 2 pool. Assuming a significant number of loads are transferred between the two pools, describe the merits of providing additional protection in the form of a cover over those storage racks directly under this frequently travelled path.

Response: The assumption that a significant number of loads will be transferred between the pools is incorrect. Both pools are nearly full which precludes significant transfers of fuel. With regard to adding a cover, this cover would only add another heavy object consideration in additon to thermal cooling concerns.

f. For both Units 1 and 2, with the aid of drawings, sequentially describe the movement of the stored spent fuel assemblies and storage racks in order to reduce the possiblity of fuel damage in the event of a load drop during the reracking operations.

Response: All work will be planned in advance and detailed procedures developed to reduce the possibility of load drops and resultant fuel damage.

g. Considering the limited space between the storage racks and the pool walls, describe the travel paths and laydown area for various pool gates. Demonstrate that the consequences of a dropped gate are acceptable or that one can reasonably assume that dropping of the gates is very unlikely.

Response: Pool gates will have to be moved over stored fuel in the new racks. Although a gate has never been dropped at Quad Cities, an analysis of such a drop of the heaviest gate from the highest potential elevation above the racks has been performed. The method and results are given in Section 7.2.5 of the Licensing Report and show 6
that no permanent deformation of the rack cells would occur.

h. Using Figure 3.7, describe and discuss the ability of the high density storage racks to protect the stored spent fuel assemblies from damage following a load drop.

Response: Two fuel assembly drop conditions are described in Section 6.6. Accident I, where the fuel assembly is postulated to drop and impact the base plate, the maximum deformation of the plate is approximately 0.5". It is proved that the base plate is not pierced. The analysis is based on a very conservative model which ignores the fluid drag of water in the cells, and does not account for material strain hardening.

Accident Condition II postulates that the fuel assembly drops on top of the rack and impacts at its weakest location. Maximum local stress in the region of impact is 22900 psi which is below the material yield point.

i. In regard to the potential for damage to stored spent fuel resulting from light load drops (i.e., one fuel assembly and its associated handling tool when dropped from its maximum carrying height), it was assumed that all lesser loads that are handled above stored spent fuel would cause less damage if dropped. Verify that this assumption was correct, e.g., indicate that all lesser loads when dropped from their maximum elevation would impart less kinetic energy upon impact with the tops of the fuel assemblies and or storage racks.

Response: Very few loads of any magnitude are permitted to be handled over the spent fuel pool. The only items transported over spent fuel are other fuel assemblies, pool canal gates, and a fuel channel measuring device. The effects of dropping these non-fuel objects onto the spent fuel storage racks were analyzed and the results reported in Section 7.2.5 of the Licensing Report. 6

12.2.4 Question:

Since Figure 2.2 shows that essentially all available space in Unit 2 pool will be occupied by storage racks, therefore, all Unit 2 stored spent fuel must be moved to Unit 1 pool via the transfer canal before it can be loaded into the shielded shipping cask. Describe and discuss what measures will be taken to reduce the possibility of fuel assembly damage resulting from the additional fuel handling operations.

Response: It will not be necessary to move all Unit 2 fuel thru the Unit 1 pool when it becomes possible to ship fuel. The racks in the Unit 2 cask handling area will not be installed unless required. If they were installed, they could be removed to facilitate the use of a cask later. In addition, all fuel movements will be accomplished by approved procedures to reduce the possibility of fuel assembly damage.

12.2.5 Question:

For both Unit 1 and Unit 2 storage pools, starting with the total decay heat load that will exist in each pool following the reracking operations, provide the following information:

a. a plot of the pool's maximum anticipated total decay heat load resulting from normal discharges versus time until each pool has reached its storage capacity.

Response: Decay heat loads for several limiting cases are discussed in Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

b. Verify that all decay heat calculations have been made in accordance with ASB technical position 9-2.

Response: All decay heat calculations have been made in accordance with Branch Technical Position APCSB 9-2 (now ASB 9-2).

c. a plot of the pool's water temperature versus time for each discharge where the total decay heat exceeds the capacity of the spent fuel pool cooling system. Indicate what cooling systems are in operation and their respective capacities.

Response: See Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

d. a plot of maximum decay heat load in each pool, assuming a full core discharge at each of the normally scheduled refueling periods.

Response: See Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

e. a plot of the pool's water temperature versus time following each full core discharge assumed in Item d above. Indicate what cooling systems are in operation and their respective capacities.

Response: See Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

f. Assuming the maximum heat load exists in Unit 1 and Unit 2 pools when all external cooling was lost, indicate the time interval before boiling occurs and the boil off rate.

Response: See Section 5 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

g. Describe and discuss the sources of makeup water, the quantity available, their respective makeup rates and the steps that must be carried out and the elapsed time before the makeup water will be available at the pools.

Response: There are 3 sources of makeup water available to the spent fuel pool. They are:

1. Using the condensate transfer pumps, water from the condensate storage tanks can be transferred to the skimmer surge tanks. There pumps can be started in minutes. Per FSAR Section 10.2-3, this system can deliver approximately 550 pgm of water cooler than that normally found in the spent fuel pool.

2. Water from the condensate storage tanks can also be transferred to the spent fuel pool utilizing the RHR pumps. This method will require the installation of a pool piece which will require about 3 hours to install. (See response to Question 12.2.2).

The amount of water available from this source is conservatively estimated to be 1000 gpm due to all flow coming into the pool via one 6 inch header.

3. River water can be delivered to the spent fuel pool within 30 minutes by use of fire hoses and one or both fire pumps. Each pump can deliver 3,200 gpm.

12.2.6 Question:

Since the RHR system will be required to augment the spent fuel cooling system for some period of time following a discharge, describe and discuss how it will be verified that the decay heat load has decayed to a value within the capacity of the spent fuel pool cooling system and, therefore, allowing the RHR system to be safely returned to its safety function mode of operation.

Response: It has been CECO's experience that the RHR is not required for either a reload or full core discharge. It was required, its use would be phased out by throttling back the RHR and observing if the pool temperature remains stable. If it is stable, the spool pieces would be removed and the RHR returned to its safety function.

12.3 <u>Questions from T. A. Ippolito to J. S. Abel transmitted on May 19, 1981</u>

12.3.1 Question:

Discuss in some detail, the procedure that will be used for (1) removal of the fuel rods from the present racks, (2) removal and disposal of the racks themselves (i.e., rating them intact or cutting and drumming them), (3) installation of the new high density racks and (4) loading them with the presently stored spent fuel rods. In this discussion include, in a step by step fashion, the number of people involved in each step of the procedure including divers if necessary, the dose rate they will be exposed to, the time spent in this radiation field and the estimated man-rem required for each step of the operation.

Response: Later

12.3.2 Question:

Demonstrate that the method used for removal and disposal of the old racks will provide ALARA exposure.

Response: Later

12.3.3 Question:

What radiation levels will be used to determine whether the racks to be disposed are identified as clean or radioactive racks.

Response: 1000 DPM per cm^2 is considered clean.

12.3.4 Question:

Identify the important radionuclides and their present concentrations (ci/cc) in the spent fuel pool water including ^{134}Cs, ^{137}Cs, ^{58}Co, and ^{60}Co. What is the external dose equivalent (DE) rate (mrem/hr) from there radionuclides. Consider these DE rates at the edge and center of the pool.

Response: See Section 8 of Supplement 2 to Revision 1 of the Licensing Report, submitted to the NRC by letter from T. J. Rausch to H. R. Denton dated August 10, 1981.

12.3.5 Question:

Provide an estimate of the increase in annual man-rem from more frequent changing of the demineralizer resin and filter cartridge.

Response: As discussed in Section 8 of Revision 1 of the Licensing Report, the proposed modification will have a negligible annual effect on the pool cleanup system; therefore, there is expected to be no increase in the annual frequency of changing of the filter demineralizer resin.

12.3.6 Question:

Discuss the build-up of crud (e.g., ^{58}Co, ^{60}Co) along with the sides of the pool and the removal methods that will be used to reduce radiation levels at the edge of the pool to ALARA.

Response: A buildup of crud as a result of this proposed modification would mean that the concentration of crud in the pool water has increased. Because the cleanup system removes essentially all crud deposited in the pool water from one refueling long before the next refueling, a measurable buildup will not occur. (See Section 8 of Revision 1 of the licensing submittal.) In addition, operating experience to date indicates no significant buildup of crud along the sides of the pool.

12.3.7 Question:

Provide an estimate of the total man-rem to be received by personnel occupying the spent fuel pool area based on all operations in that area including those resulting from 4, 5, and 6 above. Describe the impact of the modification on these estimates.

Response: As discussed in revised Section 8 in Supplement 2 of Revision 1 of the Licensing Report, there is expected to be negligible to no increase in man-rem as a result of the modification. Assuming a radiation dose of 4 mr/hr around and above the pool (see Section 8 of Supplement 2 to Revision 1 of the Licensing Report) and occupancy of 5000 man-hour during refueling and 4000 man-hour/yr at other times, the total exposures are 20 man-rem and 16 man-rem/yr respectively.

12.3.8 Question:

Identify the monitoring systems that will be used, and its location in the spent fuel pool area, that would warn personnel whenever there is an inadvertent increase in radiation levels that could trigger the alarm set-point.

Response: There are six monitoring systems with set-points of 5 mr/hr to 100 mr/hr presently monitoring the spent fuel pool area. These are deemed adequate for personnel protection.

12.3.9 Question:

I :scribe the methods used to preclude spent fuel pool water from overloading c to the spent fuel pool area floors.

Re. ponse: There are skimmers and a surge tank which will take up water displaced by the new racks.

12.3.10 Question:

Specify the present dose rate in occupied areas outside the spent fuel pool concrete shield wall and provide an estimate of the potential increase of this dose rate if the space between the spent fuel and inside concrete shield wall is reduced due to the modification.

12.3.7 Question:

Provide an estimate of the total man-rem to be received by personnel occupying the spent fuel pool area based on all operations in that area including those resulting from 4, 5, and 6 above. Describe the impact of the modification on these estimates.

Response: As discussed in revised Section 8 in Supplement 2 of Revision 1 of the Licensing Report, there is expected to be negligible to no increase in man-rem as a result of the modification. Assuming a radiation dose of 4 mr/hr around and above the pool (see Section 8 of Supplement 2 to Revision 1 of the Licensing Report) and occupancy of 5000 man-hour during refueling and 4000 man-hour/yr at other times, the total exposures are 20 man-rem and 16 man-rem/yr respectively.

12.3.8 Question:

Identify the monitoring systems that will be used, and its location in the spent fuel pool area, that would warn personnel whenever there is an inadvertent increase in radiation levels that could trigger the alarm set-point.

Response: There are six monitoring systems with set-points of 5 mr/hr to 100 mr/hr presently monitoring the spent fuel pool area. These are deemed adequate for personnel protection.

12.3.9 Question:

Describe the methods used to preclude spent fuel pool water from overloading onto the spent fuel pool area floors.

Response: There are skimmers and a surge tank which will take up water displaced by the new racks.

12.3.10 Question:

Specify the present dose rate in occupied areas outside the spent fuel pool concrete shield wall and provide an estimate of the potential increase of this dose rate if the space between the spent fuel and inside concrete shield wall is reduced due to the modification.

Response: The present (5/26/81) dose rates everywhere outside the spent fuel pool shield walls are 2 mr/hr. As seen in Figures 2.1 and 2.2 of the licensing submittal, there are at least nine inches of water between the outside of the new spent fuel racks and the thick, concrete walls of the spent fuel pool. This amount of water plus the concrete supplies sufficient attenuation that the dose rate outside the walls is negligible and changes in this dose rate due to increased spent fuel storage are not measurable. Also, there are no normally occupied spaces immediately adjacent to the concrete shield walls.

12.4 <u>Questions from T. A. Ippolito to J. S. Abel transmitted on June 16, 1981</u>

12.4.1 Question:

Describe the samples and instrument readings and the frequency of measurement that are performed to monitor the water purity and need for spent fuel pool cleanup system demineralizer resin and filter replacement. How will these be affected by the proposed action?

Response: Water purity is monitored by a continuous conductivity meter installed on the inlet to the fuel pool demineralizers, and by periodic grab samples for laboratory analysis.

Once a week a representative grab sample is obtained from the fuel pool demineralizer inlet line. The analyses performed are pH, chloride, silica, and turbidity. The activity checks are gross beta and gross alpha counts.

Once a month a sample from the same location is obtained for a gamma isotopic analysis. All major peaks are identified. All identifiable isotopes are quantified, and an LLD is determined for Kr-85.

The criteria for a demineralizer backwash and precoat is a consistent excursion from the chemistry limits, or high differential pressure across the demineralizer. Each demineralizer has differential pressure instrumentation installed which will alarm in the Unit's control room and the radwaste control room if a preset value is exceeded.

The proposed change is not expected to alter the chemistry or radiochemistry of the spent fuel pool; consequently, the described measurements will not be changed.

12.4.2 Question:

State the chemical and radiochemical limits to be used in monitoring the spent fuel pool water and initiating correcting action. Provide the basis for establishing these limits, giving consideration to conductivity, gross gamma and iodine activity, demineralizer and/or filter differential pressure, demineralizer decontamination factors, pH, and crud level.

Response: The chemical and radiochemistry limits used in monitoring the spent fuel pool water are as follows:

Conductivity	1.0 mho/cm
pH	6.0 - 7.5
Chloride	0.500 ppm
Silica	1.0 ppm
Turbidity	None
Gross Beta	1E-02 Ci/ml
Gross Alpha	1E-05 Ci/ml

If any of the above limits are exceeded the recommended action is to backwash and precoat the fuel pool demineralizer.

The basis for the water chemistry limits is the G.E. Water Quality document (22A1286, Rev. 0) that provides the water specifications for various plant systems. The limits are set to minimize corrosion and to maintain the water in a "crystal clear" condition.

The radiochemistry limits have been established based on operating experience as action levels below which personnel exposure in the vicinity of the spent fuel pools is minimized.

The demineralizers are backwashed if differential pressure exceeds 25 psid for protection of the filter elements.