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SUBJECT: Forwards response to NRC questions on 840402 application for amend to License DPR-18 re spent fuel pool.

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ROGER W. KOBER
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June 12, 1984

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Responses to NRC Staff Questions
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

Attached are responses to NRC staff questions concerning our Application for Amendment to Operating License dated April 2, 1984.

Very truly yours,

Roger W. Kober

Roger W. Kober

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1. It is stated that during normal refueling, fuel is first moved from the core to Region 1 of the spent fuel racks and not directly to Region 2. Is this an administratively or a mechanically governed procedure?

This is an administratively governed procedure. Referring to Figure 1-3 of the Application, discharged fuel from the reactor would enter the pool through the weir on the north-east side of the pool. As seen on Figure 5.4-1 of the proposed technical specification, Region 1 takes up the eastern third of the pool. To move a fuel assembly directly to Region 2, it would have to be moved over Region 1. The boundary between the two regions will be quite evident as seen from the fuel transfer bridge because of the water box configuration of Region 1. It is extremely unlikely that a fuel assembly would be placed in Region 2 through an error on the part of the bridge operator. In addition, the procedures for fuel movement are very specific during refueling and call for the transfer of discharged assemblies to specific indexed storage locations which would be designated in Region 1. These factors will prevent the placement of nonqualified fuel assemblies in Region 2.

2. Explain the derivation of the value of 0.0131 given in Table 9 as the depleted fuel assembly reactivity uncertainty.

As described in Section 2.1, page B, the value of the depleted fuel assembly reactivity uncertainty is .0102 Δk for a limiting burnup of 30,000 MWD/MT and an initial enrichment of 4.25 w/o (based on the burnup limit curve shown in Figure 23). The value of .0131 Δk is based on an earlier conservative estimate of the limiting burnup for a 4.25 w/o assembly of 40,000 MWD/MTU. From Table 10, the reactivity loss for a 4.25 w/o assembly at a burnup of 40,000 MWD/MTU is .2917 Δk/k and 5% of this value is .0146 Δk/k. The corresponding uncertainty for a calculated Region 2 multiplication factor of about .90 is .0131 Δk (.0146 x .90 = .0131). Thus, the value of .0131 Δk in Table 9 is based on a conservative estimate of the limiting burnup and was not updated based on the final limiting burnup curve shown in Figure 23.

[illegible]

3. Verify that the methods, models, and assumptions used to obtain the limit curve for Exxon and Westinghouse fuel delivered before January 1, 1984 are identical to those used to obtain the curve for the Westinghouse OFA (Figure 5.4-2 of the proposed Technical Specifications) and described in the Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack Final Report.

The methods, models and assumptions used to obtain the limit curve for Exxon and Westinghouse fuel delivered before January 1, 1984 are the same as those used to obtain the curve for the Westinghouse OFA fuel (Figure 5.4-2 of the proposed Technical Specifications). The curve for the earlier fuel is more restrictive (i.e., requires a higher burnup for a given initial enrichment) because the temperature defect (i.e., the reactivity change from cold to hot temperature conditions) is significantly larger for the earlier Exxon and Westinghouse fuel.

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4. Verify that neglecting the fact that the active fuel length (142 in.) is longer than the BORAFLEX length results in a negligible reactivity effect.

As noted in Appendix B to the Application for Amendment, additional calculations were performed to evaluate the effect of the non-full-length poison configuration. As a result of this evaluation, the poison configuration will change to that indicated on the attached figure. This figure will replace Figure 4-4 in the Application. The axial position of the poison material will bound the possible axial positions of the active fuel.

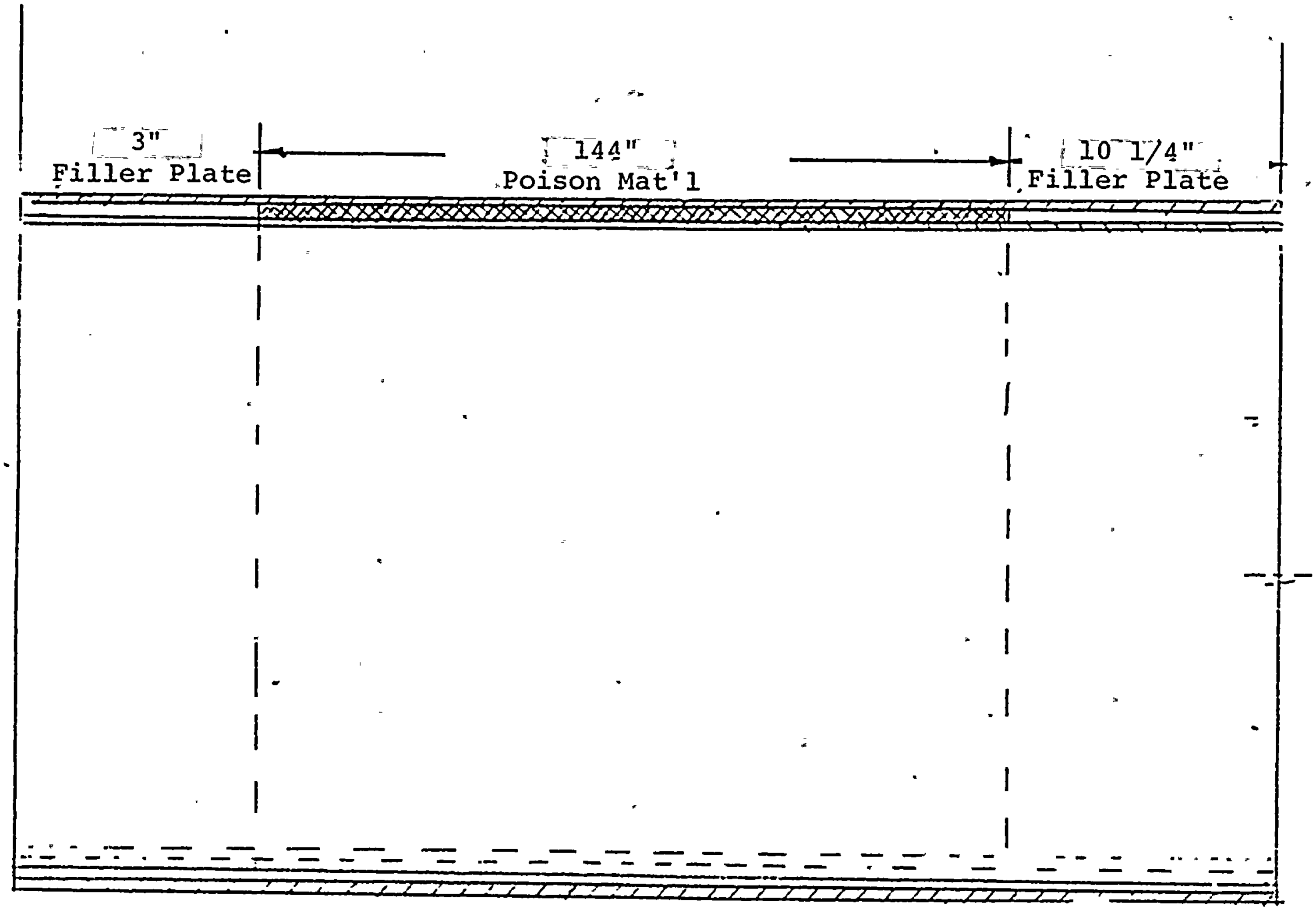

FIGURE 4-4

POISON ASSEMBLY INSTALLATION

REF. DWG 8369-5

5. Can water temperatures of less than $68^{\circ}F$ be achieved? If so, the increase in K_{∞} should be accounted for.

There have been instances during the winter months with a low heat load on the service water system where the spent fuel pool water temperature decreased below $68^{\circ}F$. We have surveyed Ginna records and have not located an instance where the temperature has fallen below $45^{\circ}F$. For Region 1 storage, the maximum rack K_{∞} decreases with decreasing pool temperature due to the effects of the flux trap rack design. For Region 2 storage, the maximum K_{∞} (including all biases and uncertainties) will increase with decreasing pool temperature. Figure 21 of the criticality analysis indicates that a change of K_{∞} with temperature is essentially linear from $100^{\circ}F$ to $68^{\circ}F$. Using the slope of this curve, the change in K_{∞} resulting from a decrease in temperature from $68^{\circ}F$ to $45^{\circ}F$ would be .00194. This would result in a maximum K_{∞} including all biases and uncertainties of .9481 which still satisfies the criteria of .95. Therefore, even under the worst case pool temperature conditions, the criteria is satisfied.

6. For multiregion spent fuel storage pools which take credit for burnup, we require (via Technical Specifications) that the procedures include an independent check of the analysis which permits storage in the burnup-dependent region(s). We also require that the records of the analysis be kept for as long as the assemblies remain in the pool.

   Attached is a change to page 5.4-3 of the proposed Technical Specification which adds the provisions requested.

associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.[3]

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The records of these calculations shall be kept for as long as fuel assemblies remain in the pool.

References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RG&E March 15, 1984.

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7. In Section 5.4, "Fuel Storage Specification," it is stated:

"In Region I it is impossible to insert fuel assemblies in other than the prescribed locations." From Figure 4-1 we note that the distance between the storage racks and the pool walls varies between 11.25 and 16.3 inches. In this regard, demonstrate that fuel assemblies cannot be inadvertently placed in these spaces and/or that the consequences are acceptable should fuel assemblies be inadvertently placed in these spaces. Should the response be that the consequences are acceptable, the analysis should account for that space where there is no fixed poison facing the pool wall, i.e., the insertion of the right-angled poison assembly in the storage cells will result in two sides of the storage racks not having poison facing the pool wall.

7. In February 1983 an application for amendment was submitted to the NRC to increase the enrichment limit for storage of unirradiated fuel in the spent fuel pool. This addressed the placement of a fresh fuel assembly adjacent to a stored assembly in the Region 1 storage configuraton. Unirradiated fuel at 4.25 w/o U-235 were assumed in both locations. Because this was a hypothetical accident condition, appropriate credit was taken for the soluble boron present in the pool water. Calculations showed that the 2000 ppm boron worth of .268 k_∞ was more than enough to compensate for the reactivity effect of the base assembly. The resulting K_∞ was .725, substantially less than the basic cell K_∞ of .9305 calculated for the normal configuration.

The abnormal configuration proposed in the question for Region 2 storage is substantially the same as that on the Region 1 - Region 2 boundary evaluated on page 10 of the criticality analysis in our submittal. The principal differences are the following.

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a. The stated question assumes the assembly was dropped at a position where two sides of the adjacent storage cell which face the pool walls do not have poison sheets. This position only occurs at the northwest corner of Region 2. While the absence of the poison sheets has a positive reactivity effect, this configuration would have greater neutron leakage providing a negative reactivity effect.

b. Because this event occurs over Region 2, the dropped assembly would be one satisfying the burnup criteria. This would provide a substantial negative reactivity effect when compared to the Region 1 - Region 2 interface configuration.

c. The negative reactivity effect of soluble boron must be taken into account.

The Region 1 - Region 2 interface K_∞ was calculated to be .9195 (.9072 + .0123). While a specific calculation for the proposed accident configuration has not been performed, it appears obvious that the negative reactivity effects of increased leakage, a depleted fuel assembly and the presencce of soluble boron would more than compensate for the absence of the poison sheets.

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8. Following the insertion of the right-angled poison assemblies, Figure 4-3 indicates the nominal interior dimensions of the storage cell will decrease from 8.280" to 8.143." In this regard, provide the following information:

   a. Indicate the minimum acceptable theoretical square space envelope inside the storage cell taking twist and bowing into account.

   b. Indicate the maximum anticipated theoretical square space envelope taking twist and bowing into account for the four spent fuel assemblies identified in Table 2-1 and the rod consolidation canister shown in Figure 4-5.

   c. What will be the maximum allowable friction forces developed during the insertion or withdrawal of the above five assemblies from the storage cells?

   d. Describe and discuss how it will be verified that the internal dimensions of the modified storage cells are within acceptable limits.

   e. Since the guide funnels and guide angles have been cut off of the storage racks, with the aid of a drawing of the four fuel assemblies and rod consolidation canister lead-ins, demonstrate that the assemblies and canister can reliably be aligned and inserted in the storage cells.

8. a. As part of the rack modification inspection plan, Westinghouse specification F-8, part B, paragraph 3 will be utilized. This states that the minimum lateral clearance and corner configuration must pass a gauge that is 52 inches long (minimum), .150 inches minimum wider than the fuel assembly, and with a .075 inch maximum corner radius or chamfer. A 100 percent check of all cells after modification will be made using a gauge that meets the requirement.

   b. The reactor design limits the fuel assembly pitch to 7.803 inches. Actual fuel assembly envelope dimension

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corresponds to approximately 7.76 inches. This is true for all fuel at Ginna. The minimum anticipated dimensions for the modified cells are less restrictive than those required in the reactor.

c. Westinghouse guidance on insertion and withdrawal drag forces specified that 50 lbs. is not to be exceeded. Westinghouse has indicated that based on experience, the forces required to damage a fuel assembly is approximately 400 lbs. We will evaluate resulting drag forces in excess of 50 lbs. on a case-by-case basis. In no case will drag forces which threaten the integrity of the fuel assembly be accepted.

d. We anticipate using a square 3 plate box gauge sized .01 inch less than the minimum dimension to check 100% of the modified cells.

e. Portable lead ins will be used to aid the operator in insertion of the fuel assemblies into the storage cells. These are currently being designed. Confidence that a suitable lead in can be developed is derived from the fact that lead ins have been developed and used on other similar designs.

None of the responses a through e address the use of rod consolidation or a canister for storage of spent fuel rods. There

is no intent to request approval of using rod consolidation or a canister for storage of spent fuel rods. The seismic analysis submitted for NRC review only incorporated the higher loadings due to rod consolidation.

9. From the statement that the pool boron concentration will be checked frequently during the storage rack decontamination process, it is inferred that unborated water will be used to decontaminate the storage racks. Describe and discuss the merits of using borated water for decontamination purposes and thereby eliminating the possibility of diluting the pool water.

9. A very minimum amount of decontamination will be attempted over the storage pool. Because of the box construction of the Ginna racks, the racks will be set down in the decontamination pit prior to the cell-by-cell decontamination process being initiated. The amount of unborated water used in spraying the outside of the racks as they are removed from the pool is negligible compared to the volume of pool water. The pool contains approximately 255,000 gallons of borated water. Assuming, for example, that 1,000 gallons were used to spray off the outside of one rack, there would only be a .4% dilution of the boron concentration of the pool water. The criticality analysis assumes unborated water in the pool. However, under certain accident conditions, credit is taken for the negative reactivity effect of the boron. Therefore, after decontamination of a rack over the pool and prior to fuel shuffling, the boron concentration will be checked.

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10. Page 25 of the submittal indicates that the previous analysis of a tipped fuel assembly drop onto a storage rack was found acceptable and implies that the same analysis is still applicable after completion of the pool expansion. In this regard, it is noted in the present proposal that the guide funnels and guide angles are to be removed. With the aid of legible drawings, discuss the results of an analysis that evaluates the consequences of a tipped fuel assembly drop into the modified storage rack filled with the most recently discharged fuel assemblies, i.e., 60 day cooling time.

10. The Ginna FSAR states on page 14.2.1-4A that a fuel assembly can be dropped 14 feet onto a flat surface and that the resulting stresses in the fuel rod were acceptably low. In response to SEP Topic IX-1 (June 9, 1981), it was determined that the resulting kinetic energy was 17,500 ft-lbs assuming all potential energy was converted to kinetic energy. Our response also determined the maximum kinetic energy of a fuel assembly plus handing tool dropped while being transferred in the pool was 16,800 ft-lbs assuming no losses due to water drag. The impact of this dropped load upon a single stored fuel assembly would result in cladding stresses in either assembly below that which Westinghouse considers acceptable. The further tipping of a dropped fuel assembly onto the stored fuel would generate a much lower amount of kinetic energy. The final loads on a stored assembly supporting the deadweight of the dropped assembly would be less than nominal design loads experienced during residence in the reactor (i.e., 900 lbs. per guide tube).

The removal of the lead ins reduces the storage box length to 158.75 in. This compares to a fuel assembly length of

approximately 160 inches. However, the full length axial support of the storage box and its high crush strength continue to protect the stored fuel from damage from dropped objects. This, in conjunction with the 60 day cooling time requirement, will insure that any possible damage to stored fuel in Region 2 will result in offsite exposures substantially less than those previously evaluated (Reference 6 of April 2, 1984 submittal).

11. Page 9 of the submittal indicates that the previously proposed and approved modifications to the cooling system for 1360 fuel assemblies would be adequate for the currently proposed pool expansion to 1016 storage cells. However, with the potential rod consolidation program, the 1016 storage cells will be capable of storing the equivalent of 1856 fuel assemblies. Therefore, describe, discuss, and demonstrate that the decay heat load from the new maximum abnormal heat load (including a full core discharge) is equal to or less than the capacity of the additional cooling loop that will be installed in 1986. Identify and discuss all differences in the assumptions made in the presently calculated heat load from the assumptions indicated in the 1981 proposed modifications.

11. As stated previously, approval for storage of consolidated fuel is not being requested in this submittal. At which time approval is requested, a thermal hydraulic analysis of consolidated fuel will be submitted. At the time the cooling system was being designed, 1360 assemblies was our estimate of the number of assemblies to be discharged through end-of-plant life.

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12. As a result of the previously expressed concern regarding the structural adequacy of the spent fuel storage pool at higher pool temperatures, a commitment was made that the decay time in the reactor vessel will increase as needed in order to assure that the decay heat loads in the pool will not exceed 16×10^6 BTU/hr, i.e., the heat removal capacity of the new cooling loop. Verify that this commitment will still be in effect following the reracking and potential rod consolidation. Quantify how this decay time may increase over the life of the facility for the maximum decay heat loads.

12. The June 9, 1981 submittal for approval of the spent fuel cooling system modifications presented results of an RG&E analysis of future decay heat loads assuming a plant life of 40 years. This analysis showed that the maximum total accumulated heat load increases from 7.07×10^6 BTU/hr in 1981 to 9.96×10^6 BTU/hr in the year 2010. Assuming a full core discharge in 2010, the maximum rated capacity of the new system of 16×10^6 BTU/hr will not be exceeded by using an in reactor cooldown time of 14 days. This new system will be installed in 1986. As stated in the submittal, the current cooling system has sufficient capacity for normal discharges through 1986. We do not anticipate a full core discharge prior installation of the new cooling system.

13. In regard to the cruciform bottom plates indicated in Figure 4-7 at the four corners of the storage racks, with the aid of legible working drawings, provide a discussion that demonstrates that the cooling flow resistance to the corner cells is essentially equal to that of all other storage cells and, therefore, the temperature of the water exiting from the corner cells is not significantly higher than that from other cells.

13. Figure 4-7 of the submittal does not show the 3 1/2" holes in the cruciform to allow coolant flow to the cells. These holes are essentially the same as those in the other storage cells. Holes in the cruciform are aligned with holes in the support base to allow for adequate flow to these cells. The support base does not provide an obstruction to flow to any storage cell in the modified racks. It has been previously calculated that the peak cladding temperature for the peak assembly of the hot batch stored farthest from the cooling system discharge would be 159.1°F.* This is for fuel stored in the Region 1 configuration at a position which insures the maximum resistance to flow.

*NRC Safety Evaluation of Spent Fuel Storage Rack Replacement, November 15, 1976.

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