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U. S. NUCLEAR REGULATORY COMMISSION
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Gentlemen:

DOCKETS 50-266 AND 50-301
TECHNICAL SPECIFICATION CHANGE REQUEST 124
NUCLEAR FUEL STORAGE ENRICHMENTS
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the requirements of 10 CFR 50.59(c), Wisconsin Electric Power Company (Licensee) hereby requests amendments to Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Plant, Units 1 and 2, respectively, to incorporate a change in the plant Technical Specifications. The proposed change, which is discussed in detail below, provides for storage of fuel assemblies of higher enrichment than is currently specified at Point Beach.

The change is identified with a margin bar on the attached proposed Technical Specification page (Attachment 1). The proposed change is to Specification 15.5.4.2 and would increase the grams of U-235 per axial centimeter limit for Westinghouse OFA fuel assemblies from 39.4 to 46.8 and would permit use of OFA fuel with or without axial "blankets" of lower enrichment fuel material. This increase in the grams of U-235 per axial centimeter corresponds to an increase in fresh fuel enrichment from the current limit of 4.0 to 4.75 weight percent U-235 (w/o).

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Criticality analyses of the Point Beach new and spent fuel storage racks for storage of Westinghouse OFA fuel was originally performed by Pickard, Lowe and Garrick, Inc., (PLG) in support of Technical Specification Change Request 87 (see Attachment C to our letter to Mr. H. R. Denton dated September 6, 1983). Technical Specification Change Request 87 addressed all aspects of OFA fuel utilization and included storage of OFA fuel at enrichments of up to 4.0 w/o (i.e., 39.4 grams per axial centimeter). This change request was approved by the NRC, resulting in Amendments 86 and 40 to Facility Operating Licenses DPR-24 and DPR-27, respectively, dated October 5, 1984.

The original PLG analyses were performed using analytical techniques similar to those used for licensing other plants, as well as for the 1979 licensing of the high-density spent fuel storage racks at Point Beach. These criticality analyses utilized the LEOPARD and PDQ-7 computer programs. LEOPARD and PDQ-7 calculational accuracies were verified by means of benchmark comparisons with critical assembly experiments, and conservative techniques were used for the determination of the infinite neutron multiplication factor, k-infinity. The original PLG analyses actually supported storage of 4.75 w/o, but only 4.00 w/o was licensed because, at that time, use of enrichments greater than 4.0 w/o was not contemplated. This is demonstrated in Attachment 2, which presents Figure 2 of the original study performed in support of Technical Specification Change Request 87.

PLG has reviewed its earlier analyses in light of current requirements and concludes that the earlier analyses do, indeed, support storage of OFA fuel at enrichments of 4.75 w/o in the Point Beach spent fuel pool under the current limitation of maintaining a 5% shut-down margin (k_{eff} <0.95, as stipulated in TS 15.5.4.2). It should be noted that all specified and calculated neutron multiplication factors assume a pool with unborated water. The maximum k-infinity for the spent fuel rack (not including biases and uncertainties) with fuel at 4.00 w/o was determined to be 0.8993, as may be noted from Attachment 3 (Table 5 of the original analyses report). Attachment 3 also lists the calculational biases, tolerances, and uncertainties associated with the analyses. Applying these conservative adjustments, the maximum k-infinity for the spent fuel rack with fuel at 4.00 w/o is determined to be 0.9100.

A similar approach is taken to determine k-infinity for higher enrichment fuel. Inspection of Attachment 2 reveals the k-infinity for fuel of the same design (not including biases and uncertainties), but at an enrichment of 4.75 w/o, is 0.9295. The resulting delta k/k associated with this increase in enrichment is 0.0336, i.e., (0.9295-0.8993/0.8993). The maximum k-infinity for the spent fuel racks, including all biases and uncertainties, with fuel at an enrichment of 4.75 w/o U-235 is, therefore, calculated to be 0.9406, i.e., 0.9100 x (1.0000 + 0.0336). This leaves an adequate margin to the limiting licensing value of 0.95 (k_{eff}).

PLG has also reanalyzed the new fuel vault storage racks in the flooded state. In addition to the base case run for a flooded cavity, cases were run at elevated temperatures and for mist conditions, with water densities ranging from 3% to 80% of maximum water density. A case was also run in which the fuel assembly is off-center in the unit cell, and another case was run for the maximum pellet density. All runs were performed for Westinghouse 14 x 14 OFA fuel at an enrichment of 5.50 w/o U-235. The calculational biases and uncertainties were derived from those used in the original report, except that the benchmark cases using thin absorbers were excluded. The LEOPARD/PDQ model bias in this case is 0.0071 ΔK, and the calculational uncertainty is 0.0029 ΔK. The combination of the biases and uncertainties is shown in Attachment 4 (Table 1 of the recently completed analyses). The maximum expected value of k-infinity for the new fuel storage racks with 5.50 w/o fuel, including biases and uncertainties, is 0.9221, which is lower than the regulatory limit of 0.95 (k_{eff}). Based on these analyses and on the spent fuel analyses, the spent fuel storage racks are seen to be the "limiting case".

While we presently have no plans to utilize an axial-zoned core-loading scheme, this option is a future possibility. In order to justify the use of axially zoned OFA fuel containing lower enrichment uranium at the top and bottom of the fuel, PLG reviewed the effect of these axial "blankets" on the fuel rack multiplication factors. As might be expected, replacing any fuel in an assembly with fuel of lower enrichment will reduce the assembly multiplication factor, k-infinity. Specifying the limit in terms of grams of U-235 per axial centimeter ensures that the limiting multiplication factor will not be exceeded.

In all cases the use of lower enrichment uranium ends will increase the subcriticality margin of the new or spent fuel storage racks beyond that previously calculated for the new or spent fuel racks. Since the spent fuel pool is the limiting case, only fresh fuel of less than, or equal to, 4.75 w/o (46.8 grams U-235 per axial centimeter) will be used at Point Beach under the proposed Technical Specification change.

Attachment C to the September 6, 1983, transmittal regarding Technical Specification Change Request 87 also addressed the transition to OFA fuel with respect to spent fuel cooling, radiological effects, and gamma heating effects on the spent fuel racks, poison material, and spent fuel pool walls. Because using higher enrichment fuel will be done in conjunction with increasing discharge burnups, the effects of higher discharge burnup on these aspects of spent fuel storage were also reviewed. Presently, we are designing OFA cores for a discharge burnup of approximately 40,000 MWD/T.

Our plans call for using a modified version of the Westinghouse OFA fuel, at 4.75 w/o U-235, which will have the same neutronic characteristics as the current OFA fuel. The only mechanical changes to this fuel relate to the top and bottom nozzle configurations and the increased fission gas plenum volume in the fuel rods. Average discharge burnups of 45,000 MWD/T are planned for cores utilizing the new fuel. (Westinghouse has justified fuel performance at much higher burnups in its document WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel", P. J. Kersting, et. al., July 1982. This topical report was accepted by the NRC in October 1985).

We recently completed a fuel analysis utilizing the ORIGEN computer code with the intent of quantifying the effects of increased discharge burnups (for fuel with a higher initial enrichment of 4.75 w/o) relative to decay heat, radiation doses, and curie content of the fuel. ORIGEN - the Oak Ridge National Laboratory (ORNL) Isotope Generation and Depletion Code - is a versatile point depletion code which solves the equations of radioactive growth and decay for large numbers of isotopes and can be used to compute the compositions and radioactivity of fission products, cladding materials and fuels. As expected, our analysis showed little dependence of those parameters on initial enrichment. In fact, decay heat, gamma dose and curie content are slightly less if the initial enrichment is increased from 4.0 w/o to 4.75 w/o. Regarding the effects of increased discharge burnup, similar conservative

results are also seen. For approximately the first three years after discharge, the parameters are reduced. Beyond three years, for a typical fuel assembly, the opposite is true. Decay heat generation, gamma doses and curie content are greater than that seen at lower burnups, being roughly in proportion to the increase in burnup. However, over the long term for the entire spent fuel pool, these parameters are roughly the same as they were at lower burnup because higher discharge burnups result in fewer fuel assemblies being discharged, on the average, per cycle.

The ORIGEN calculations also confirm that heating and gamma dose considerations are bounded by the original OFA analysis which covered two cases: a normal refueling load (1502 spent fuel assemblies) and the core unload (1381 spent fuel assemblies plus 121 assemblies [after 30 days at power]).

As required by 10 CFR 50.91(a), we have evaluated this change in accordance with the standards specified in 10 CFR 50.92 to determine if the proposed change constitutes a significant hazards consideration. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

Regarding the first criterion, a higher U-235 enrichment does not affect any accident previously evaluated. This parameter is not considered in any accident analysis performed for the Point Beach facility. By the same reasoning, the second criterion is also addressed. A change in fuel enrichment, in itself, cannot cause a new or different kind of accident.

The third criterion, margin of safety, must be addressed. The area of concern surrounding a change in fuel enrichment is the effect on the neutron multiplication factor, both in the spent fuel and new fuel storage areas of the spent fuel pool. The margin of safety, from a standpoint of criticality, is established in the Technical Specifications. TS 15.5.4.2 clearly defines that margin by specifying that, for both the spent fuel and new fuel storage locations, k_{eff} shall be less than 0.95 with the storage pool filled with unborated water.

(Note that the specifying of unborated water is for analysis only, and with the intent to build in additional conservatism to the specification. TS 15.5.4.3 stipulates the minimum boron concentration for the spent fuel pool as 1800 ppm whenever spent fuel is being stored.) We have established that the neutron multiplication factor, with the additional, conservative assumption of fuel stored in an infinite array (k-infinity), is less than the specified limit of 0.95 (k_{eff}) for new and spent fuel of an initial enrichment of 4.75 w/o U-235. Margin of safety is, therefore, not reduced beyond the previously specified value.

Secondly, we have addressed the case of fuel assemblies containing low enrichment uranium at the top and bottom of the fuel. While we presently have no plans to utilize this "axial blanket" core loading at Point Beach, the storage of such fuel in the spent fuel pool is not a concern in that replacement of any amount of fuel with fuel of lower enrichment will reduce the multiplication factor, provided the enrichment of the central section of the fuel is not increased to compensate. In specifying enrichment in terms of grams U-235 per axial centimeter, this compensatory loading of fuel rod central sections above 4.75 w/o U-235 is precluded. By this specification, storing this type of fuel can only increase the margin of safety presently established.

Lastly, the influence of higher enrichment fuel on spent fuel pool cooling, radiological effects, and gamma heating effects was addressed. The original OFA analysis in support of Technical Specification Change Request 87 quantified those effects for spent fuel of initial enrichment 4.0 w/o U-235. We have completed an analysis which demonstrates conservative changes in the parameters of interest for this higher enrichment fuel at increased burnups. We conclude that the original analysis completed for OFA fuel, and found acceptable by the NRC in the Safety Evaluation in support of your October 5, 1984, amendment transmittal letter, bounds the expected effects resulting from the proposed change. Margin of safety is, therefore, not reduced beyond the previously specified value.

We conclude by this evaluation that the proposed specification does not involve a significant hazards consideration.

We have enclosed a check in the amount of $150 for the application fee as prescribed in 10 CFR 170. Please contact us if you have any questions regarding this submittal.

Very truly yours,

C W Fay

C. W. Fay
Vice President
Nuclear Power

Enclosures (Check 633405)

Copies to NRC Regional Administrator, Region III
NRC Resident Inspector
R. S. Cullen, PSCW

Subscribed and sworn to before me
this 6th day of July, 1988.

Delores B Gryczkowski
Notary Public, State of Wisconsin

My Commission expires 5-27-90.