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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

INTRODUCTION

By letter dated March 14, 1983, Wisconsin Electric Power Company (the licensee) made application to amend the Technical Specifications (TS) of Point Beach Nuclear Plant, Units 1 and 2, to allow use of Westinghouse Optimized Fuel Assemblies. Several other core parameter changes were also requested. A safety analysis supporting the application was submitted by letter dated September 6, 1983, along with some revised TS changes. Further information in response to staff questions was submitted by letters dated July 13 and August 17, 1984. The July 13 submittal also provided relabeling of the axis on two figures from the previous request. The staff has reviewed the application and prepared the following evaluation.

DISCUSSION AND EVALUATION

The proposed changes to the Point Beach Technical Specifications are required in order to accommodate a change in the fuel design from Westinghouse standard 14x14 fuel assemblies to the Westinghouse 14x14 optimized fuel assembly (OFA). In addition, the analysis and operating procedures for the reactors will be altered to include the following:

1. A change in the power dependent term in the $F_{\Delta H}$ limit algorithm from 0.2 to 0.3.

2. Use of the Relaxed Axial Offset Control (RAOC) strategy instead of the current Constant Axial Offset Control (CAOC) strategy.

3. Use of 0.95 for the value of the refueling k-effective instead of the current 0.90.

4. Allowance for a positive moderator coefficient below 70 percent of full power.

5. Use of the Westinghouse Improved Thermal Design Procedure (ITPD) for the OFA fuel along with the WRB-1 DNB correlation.

6. Increase of scram insertion time from 1.8 to 2.2 seconds.

Incorporation of the changes required an examination of all the transients and accidents and partial or complete reanalysis of many of them. A discussion of the effect of the various proposed changes follows.

1. Fuel Mechanical Design

The most significant differences between the new OFA fuel and the current standard fuel are the smaller diameter of the fuel rods and guide tubes in the OFA design and the replacement of the Inconel inner grids of the standard design with Zircaloy grids in the OFA design. The effect of these changes on the mechanical performance of the fuel has been considered.

The Zircaloy grids are somewhat wider and thicker than those they replace. The additional thickness means that the diameter of the instrument and guide tubes must be reduced. The control rod insertion time assumed for safety analyses has been increased from 1.8 to 2.2 seconds to account for the reduced guide tube diameter. The Zircaloy grids are located in the axial locations compatible with those in the standard fuel assemblies. The greater width and thickness of the interior grids has implication for the hydraulic compatibility of the two fuels as discussed in Section 3 below.

The mechanical design requirements criteria which have been approved for the 17x17 OFA design are met for the OFA design. These include the prevention of cladding collapse during the design lifetimes of the fuel rod, the limiting of internal gas pressure to preclude outward cladding creep during steady state operation, acceptable grid deformation during seismic or LOCA events and acceptable fretting wear due to flow induced vibration.

2. Nuclear Evaluation

The nuclear evaluation of the transition and all-OFA cores has been performed with the Westinghouse Reload Safety Methodology which has been used in previous Point Beach reload analyses.

The results show that the expected values of most of the nuclear parameters fall within the normal cycle-to-cycle variations. A notable exception is the moderator temperature coefficient which is positive at low power as a result of the presence of the OFA fuel.

The use of the Relaxed Axial Offset Control (RAOC) strategy instead of Constant Axial Offset Control (CAOC) required that a different set of analyses (xenon transients) be performed. The procedures used were those that have been approved for obtaining ROAC operating limits. The limits were established to satisfy the peaking factor constraints imposed by LOCA analysis.

On the basis that the nuclear evaluation has been performed with previously accepted methods, the staff concludes that it is acceptable.

3. Thermal-Hydraulic Evaluation

The presence of transitional mixed cores containing both standard and OFA fuel requires that particular attention be paid to the thermal-hydraulic analysis of the core. Hydraulic compatibility of the two fuel types was established by a series of tests in the Westinghouse Fuel Assembly Test System facility.

Different DNBR correlations are used for the two fuel types. For the standard fuel type the W-3 correlation with a design limit DNBR of 1.3 is used. This includes a generic margin of 18.1 percent which is used to offset rod bow and mixed core effects. For the OFA fuel the WRB-1 DNB correlation is used with the "Improved Thermal Design Procedure" and the THINC IV computer code. Use of this correlation of OFA fuel has been demonstrated and documented in WCAP-9401-A for 17x17 fuel.

Confirmatory tests have been performed for 14x14 OFA fuel to verify that the WRB-1 correlation with a design DNBR limit of 1.17 is appropriate.

In the Improved Thermal Design Procedure, the safety analyses are performed using nominal values of the plant operating, nuclear, thermal, and fuel fabrication parameters. Uncertainties in the DNBR value due to variations in these parameters are combined statistically and added to the DNBR design value (1.17) to obtain a target value. The values obtained for this quantity for Point Beach are 1.32 for thimble cells and 1.33 for typical cells. The licensee has provided information concerning the plant specific uncertainties for Point Beach which support these values. Transition core and rod bow effects are not included in the target values. In order to account for these effects, additional margin is provided to arrive at analysis values which are 1.65 and 1.66 for thimble and typical cells, respectively.

The fractional closure due to rod bow has been estimated to be the same for the two types of fuel. The rod bow would be increased for the OFA fuel relative to the standard fuel but the rod-to-rod gap is greater. Thus it is concluded that the same rod bow penalty may be used for both fuel types. Since large DNBR margins exist to account for the penalty, this is acceptable.

A transition core DNB penalty of one percent has been determined to be applicable to both types of fuel when they are together in a mixed core. This determination was made by performing analyses with different core loading patterns at various core conditions in a manner consistent with that previously used (the R. E. Ginna Cycle 14 reload, e.g.) and approved.

Fuel temperatures for use in the safety analyses were calculated with the PAD fuel performance code with conservative inputs for certain key parameters. This procedure has been previously used for this purpose and is acceptable.

4. Transient and Accident Analyses

An extensive re-evaluation of the transient and accident analyses for the Point Beach reactors was performed to address the following changes:

1. Optimized Fuel
2. Positive Moderator Temperature Coefficient
3. $F_{\Delta H}$ Multiplier Change
4. Relaxed Constant Axial Offset Control
5. Rod Drop Time Increase
6. Refueling Shutdown Margin Decrease

Each of the transient and accident events was examined to determine whether any of the changes listed above would affect its consequences in an adverse manner. For those events that had altered consequences, a new analysis was performed. Of the events examined, only 3 - startup of an inactive coolant loop, loss of normal feedwater, and loss of all AC power to the station auxiliaries - were found to be unaffected. The rest were affected by one or more of the proposed changes. All of the reanalyses used the increased scram time though this affected only the Rod Ejection Accident, startup accident and the loss of coolant flow. For all reanalyses, the DNBR evaluation was performed separately for the OFA and Standard fuel as described in Section 4 above.

The change in the $F^N_{\Delta H}$ multiplier results in larger permissible values of $F^N_{\Delta H}$ at lower powers. This may impact the axial offset envelop such that the f (ΔI) term in the protection circuitry changes. However, no credit is taken for this term in safety analyses and the change in the multiplier has no impact on the analyses. The reduction in Refueling Shutdown Margin impacts only the Boron Dilution Accident at Refueling Conditions.

The accident evaluations and analyses were performed to encompass both Point Beach Units 1 and 2. The following analyses were performed:

° Types of Core
  Full Standard Core
  Transition Core
  Full OFA Core

° Operating Pressures
  Normal Pressure of 2250 PSIA
  Reduced Pressure of 2000 PSIA

° Steam Generators
  Unit 1 - Model 44F with 11% effective plugging
  Unit 2 - Model 44 with 14% effective plugging

The analyses were performed for each core type and bounding results were used so as to obviate the need for Technical Specification changes in succeeding cycles. DNB limiting transients were analyzed at 2000 PSIA pressure and overpressure transients at the higher pressure. The most conservative results from the two units were used to establish Technical Specification limits so that a single set of Technical Specifications might be used for both Units 1 and 2.

For most reanalyzed events, large margins exist between the minimum DNBR values reached during the transient and the established analysis limits (1.65 and 1.66). The limiting DNBR event for the Point Beach reactors is the Uncontrolled Rod Bank Withdrawal at Power. The minimum DNBR obtained from the reanalysis of this event showed essentially the same margin to the analysis limits (~0.05 in DNBR) as did the previous analysis.

Protection against overpressurization events is provided by safety valves whose settings have not changed. Sufficient capacity is provided to preclude significant changes in peak pressure for the reanalyzed events. The Boron Dilution event during refueling was reanalyzed to account for the reduced refueling shutdown margins.

.he revised analysis shows that 52 minutes are required to reach criticality after the onset of dilution. Source range monitors will provide an alarm at least 15 minutes before criticality. The staff concludes that sufficient time exists to allow the operator to take action to preclude criticality. The Boron Dilution event at cold shutdown was also reanalyzed to account for the presence of the OFA fuel.

The required shutdown margin as a function of Boron concentration was revised to assure that at least 15 minutes are required to achieve criticality. This is the same criterion as previously employed and is acceptable.

Both the small-break and large-break LOCA events were reanalyzed for the limiting all-OFA fuel core. The currently accepted models were used for both events. The large-break event resulted in a requirement for a full power F_Q value of 2.21. Small changes were also required in the K (z) curve. The RAOC analyses were performed within the constraints imposed by the new requirements.

The small-break LOCA analysis resulted in peak clad temperature of less than 1000°F with assumed F_Q values of 2.32 at core center and 1.5 at the top of the core. This is far below the acceptance criterion of 2200°F and is acceptable.

In response to a staff question, the licensee provided an analysis of the dropped rod event for very small rod worths. This had been identified by Westinghouse as a possible non-conservatism in the FSAR analysis. The new analysis showed that the DNBR criterion was not violated. This is acceptable.

5. Technical Specifications

The staff has reviewed the Proposed Technical Specifications and finds them acceptable. This conclusion is based on the following:

1. The Specifications are consistent with the assumptions used in the safety analyses (Specifications 15.1.g, 15.3.1.F, 15.3.1.G, 15.3.10.B ($F_{\Delta H}$), 15.3.10.D, and 15.3.10.E),

2. The Specifications are consistent with the results of the safety analysis (Specification 15.2.1, 15.2.3, 15.3.10.B (F_Q) Figures 15.3.10-1, 15.3.10-3, and 15.3.10-4),

3. The Specifications are descriptive in nature (Specifications 15.5.3.A and 15.5.4), and

4. The Specifications provide clarification (Specification 15.3.6.A and Table 15.4.1-1)

The changes to the bases for the various Technical Specifications, which have been revised to make them consistent with the Specification, are also acceptable.

Based on our review which is described above, the staff concludes that the proposed changes to the Technical Specifications are acceptable. This conclusion is based on the following:

1. The analysis methods used have been previously reviewed and accepted.

2. The analyses properly account for the changes in core design and operation.

3. The consequences of the revised analyses show insignificant reductions to previous operating margins.

6. Analysis of Storage of OFA Fuel at Point Beach

The use of optimized fuel in the Point Beach reactors requires the reevaluation of the fresh and spent fuel storage facilities. The effect of the new fuel on criticality, spent fuel cooling requirements, radiological consequences and gamma heating effects were examined. The k-effective of the fresh fuel in spent fuel storage racks was calculated assuming storage of OFA fuel with 4 weight percent U-235. The calculations were performed with the same methods that had been previously used and approved for the racks. For the fresh fuel racks, the k-effective value, including uncertainties, was 0.872 for fully flooded rack and 0.894 for the low density moderation yielding the highest value. For the spent fuel racks, the k-effective value, including uncertainties, was 0.910. These values meet our acceptance criteria for the racks and are therefore acceptable.

The effect of the new fuel on the spent fuel cooling requirements is negligible compared to the 10 percent uncertainty assumed in the calculations. The smaller rod diameter results in more direct gamma heating of the coolant but the increase is not enough to lead to boiling in the water between storage locations. The additional leakage of gamma radiation into the pool water does not lead to increased exposure at the pool surface since the large depth of water over the fuel attenuates the dose by a large factor. The increased gamma leakage from the fuel results in about a ten percent increase in the total dose to the poison material (Boraflex). The poison surveillance program will be altered to cover the increased exposure.

The staff concludes that the storage of OFA fuel in the fresh and spent fuel storage racks meets the staff requirements for such storage and is acceptable.

Environmental Consideration

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

Conclusion

We have concluded, based on the considerations discussed above, 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 5, 1984

Principal Contributors:
Walter Brooks
T. G. Colburn