ATTACHMENT B

Safety Evaluation
for
Point Beach Units 1 and 2
Transition to
Westinghouse 14x14 Optimized Fuel Assemblies

Edited by

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Wisconsin Electric Power Company

Point Beach Units 1 and 2

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1.0 INTRODUCTION

Point Beach Units 1 and 2 have been operating with all low-parasitic standard (STD) fueled cores. It is planned to refuel both Units 1 and 2 with optimized fuel assembly (OFA) regions. As a result, future core loadings would range from approximately 1/3 OFA plus 2/3 STD cores to eventually an all OFA fueled core. The 14x14 OFA fuel has similar design features compared to the STD Westinghouse 14x14 fuel which has had substantial operating experience in a number of nuclear plants. The major differences are the use of 5 intermediate (mixing vane) Zircaloy grids for the OFA fuel versus 5 intermediate (mixing vane) Inconel grids for the STD fuel, and a reduction in the fuel rod diameter. (See Table 1-1).

This report summarizes the safety evaluation performed on the region-by-region reload transition from the present Point Beach Units 1 and 2 STD fueled cores to cores with all OFA fuel. This report examines the differences between the STD Westinghouse fuel assembly designs and evaluates the effects of these differences for the transition to an all OFA core. The evaluation considers the STD reload design methods described in Reference 1, and the transition effects described in Chapter 18 of Reference 2.

Reference 3 presents the operating experience from the four 14x14 demonstration assemblies loaded in Unit 2. The purpose of this program was to obtain early performance information on the OFA design to confirm its adequacy prior to insertion into Unit 2 in late 1984. The four demonstration assemblies have completed two cycles of operation (established burnup ~20,000 MWD/MTU). Post-test examination at the completion of the first cycle of irradiation indicated no abnormalities. However, one demonstration assembly at the end of the second cycle of irradiation was damaged and removed. It was concluded that the cause of the damage was an isolated event and not a generic OFA design problem (see Letter Report IT-83-222, "Failure Investigation of Point Beach Unit 2 OFA Rods", July 1983). The demonstration assemblies will have experienced approximately 35,000 MWD/MTU of burnup prior to the late 1984 OFA loading date for Unit 2.

Sections 3 through 6 summarize the Mechanical, Thermal and Hydraulic, Nuclear, and Accident evaluations, respectively.

TABLE 1-1

COMPARISON OF 14x14 OFA AND 14x14 STD ASSEMBLY DESIGN PARAMETERS

PARAMETER	14x14 OFA FUEL ASSEMBLY DESIGN	14x14 STD FUEL ASSEMBLY DESIGN
Fuel Assy Length, in	159.710	159.710
Fuel Rod Length, in	151.850	151.850
Assembly Envelope, in	7.763	7.763
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in	.556	.556
Number of Fuel Rods/Assy	179	179
Number of Guide Thimbles/Assy	16	16
Number of Instrumentation Tube/Assy	1	1
Compatible with Movable Incore Detector System	Yes	Yes
Fuel Tube Material	Zircaloy 4	Zircaloy 4
Fuel Rod Clad OD, in*	0.400	0.422
Fuel Rod Clad Thickness, in	0.0243	0.0243
Fuel/Clad Gap, mil*	7.0	7.5
Fuel Pellet dia, in*	0.3444	0.3659
Guide Thimble Material	Zircaloy 4	Zircaloy 4
Guide Thimble OD, in*	0.526	0.539
Guide Thimble Wall Thickness, in	0.017	0.017
Structural Mat'l - Five Inner Grids*	Zircaloy	Inconel
Structural Mat'l - Two End Grids	Inconel	Inconel

*Note: OFA design change compared to STD fuel assembly

2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse STD reload methodology for analyzing cycle specific reloads (Reference 1), parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate subsequent determination of the applicability of 10 CFR 50.59. The objective of subsequent cycle specific reload safety evaluations will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this report. The mechanical, thermal and hydraulic, nuclear, and accident evaluations considered the transition core effects described for mixed cores in Chapter 18 of Reference 2. The summary of these evaluations for the Point Beach Units 1 and 2 transitions to an all OFA core are given in the following sections of this submittal.

The transition design and safety evaluations consider the nominal conditions which are consistent with 100% of the FSAR thermal design flow, Reference 5. These nominal conditions are 1518.5 MWt core power, 2000 or 2250 psia system pressure, core inlet temperatures of 545.0°F at 2000 psia, 545.3°F at 2250 psia and 178,000 gpm RCS thermal design flow.

The results of evaluation/analyses and tests discussed in this report lead to the following conclusions:

1. The OFA reload fuel assemblies for the Point Beach Units 1 and 2 are designed to be mechanically compatible with the current STD 14x14 fuel assemblies, control rods, and reactor internals interfaces. Both fuel assemblies satisfy the current design bases for the Point Beach Units.

2. Generally, changes in the nuclear characteristics because of the transition from STD to OFA fuel will lie within the cycle-to-cycle variations observed for past STD fuel reload designs. The moderator temperature coefficient is the most significant exception to this.

Since the H/U ratio is larger for OFA, the moderator temperature coefficient is more positive than observed in past STD fuel Point Beach cores. This has been accounted for in the accident evaluations.

3. The reload OFAs are hydraulically compatible with the current STD reload assemblies.

4. The accident analyses for the OFA transition core were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature, as required. The previously reviewed and licensed safety limits are met. Analyses in support of this safety evaluation establish a reference design on which subsequent reload safety evaluations involving OFA reloads can be based.

5. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 7 of this report.

3.0 MECHANICAL EVALUATION

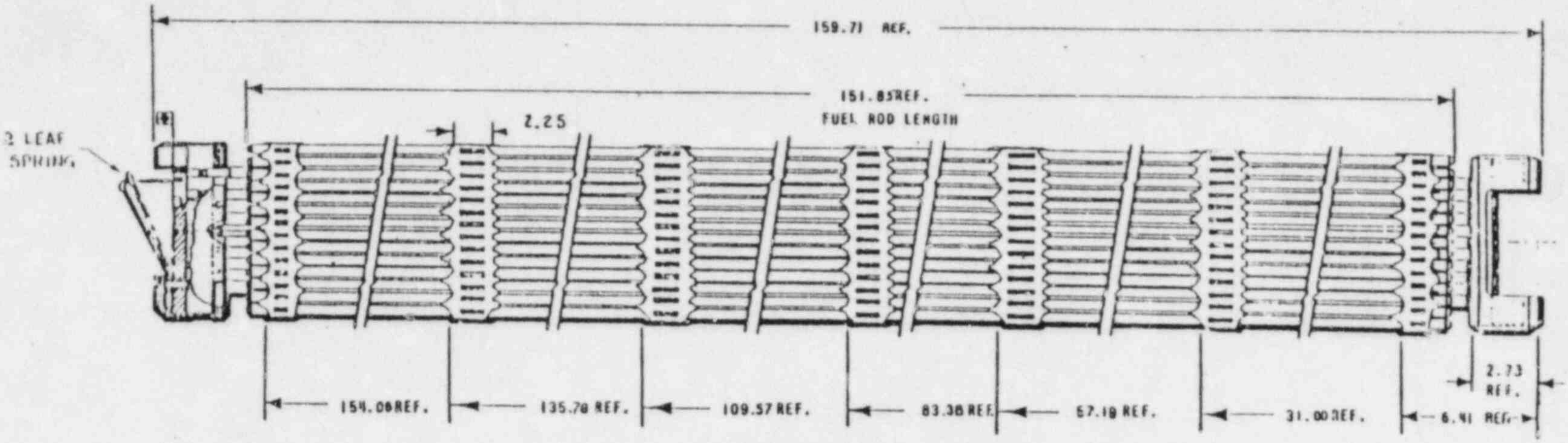
The mechanical design requirements and criteria approved by the NRC for the 17x17 OFA design are described in Reference 2. The 14x14 OFA design meets these same basic design requirements and criteria.

The 14x14 OFAs have been designed to be mechanically compatible with the STD fuel assemblies, reactor internals interfaces, fuel handling and refueling equipment. Figure 3-1 presents a comparison of the OFA and STD fuel assemblies. The grid elevations for the two assembly designs match, minimizing mechanical and hydraulic interaction. The assembly envelopes, fuel rod design/dimensions, and top and bottom Inconel grids are the same. Some basic changes between the STD and the OFA fuel assembly design include: a reduction in guide thimble and instrument tube diameters, replacement of the five intermediate Inconel grids with Zircaloy grids, a reduction in fuel rod diameters, and the implementation of a modified bottom nozzle to facilitate reconstitution (fuel assembly repair).

The OFA fuel rod will maintain the same overall length as the STD fuel rod. However, the fuel rod diameters are reduced to optimize the water to uranium ratio.

The top and bottom Inconel grids of the OFA are nearly identical in design as the Inconel grids of the STD fuel assembly. The five intermediate grids will be made of Zircaloy material rather than of Inconel which is currently used in the STD design. The Zircaloy grids have thicker straps than the Inconel. Also the Zircaloy grid height is approximately .75 inch greater than the Inconel grid height. Dimensional changes are mainly due to a difference in material structural properties. Due to this difference in material structural properties, dimensions were increased in order to maintain approximately the same grid strength for the OFA Zircaloy grid as for the original STD inconel grid. The Zircaloy grid incorporates the same grid cell support design as the Inconel.

RECONSTITUTABLE OPTIMIZED FUEL ASSEMBLY 14x14



STANDARD FUEL ASSEMBLY 14x14

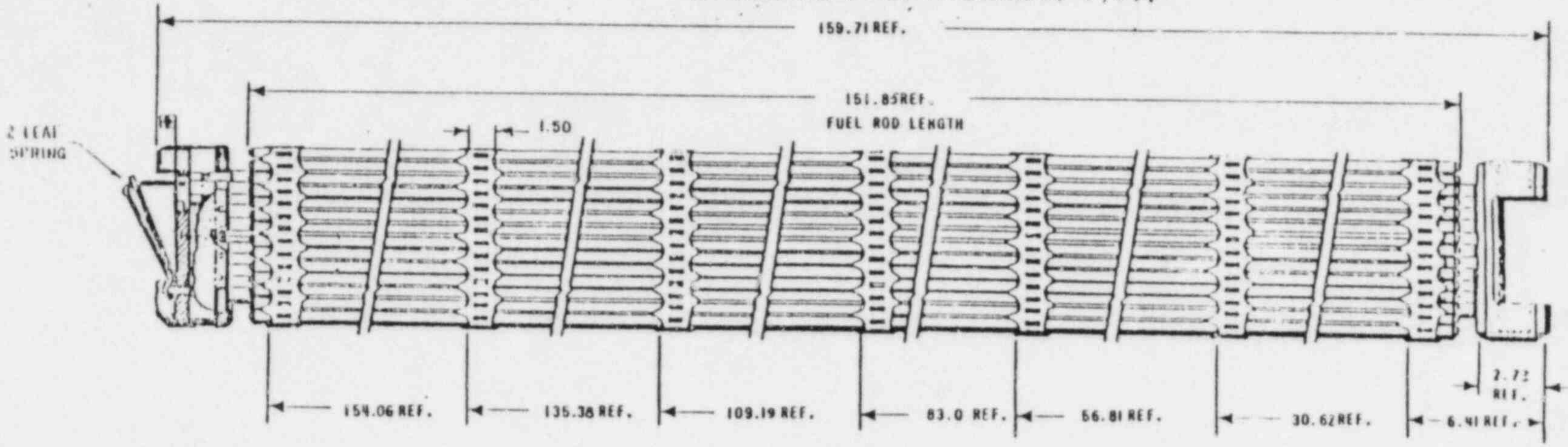


FIGURE 3-1
RECONSTITUTABLE OFA AND
STD 14x14 FUEL ASSEMBLY

Due to the thicker Zircaloy grid straps and the resulting reduced cell size, the OFA guide thimble tube ID (above the dashpot) has a 13 mil reduction when compared to the STD thimble tube ID. The OFA guide thimble tube ID provides an adequate nominal diametral clearance of 61 mils for the control rods. However, due to the reduced clearance compared to the STD assembly thimbles, the scram time for accident analyses is increased from 1.8 secs (to dashpot) for the STD assembly to 2.2 secs for the OFA. The new design continues to provide an adequate nominal diametral clearance for control rods as well as other core components.

The 14x14 OFA top nozzle is identical to the 14x14 STD top nozzle. The 2-leaf holdown springs designed for the 14x14 STD top nozzle continue to meet all applicable design criteria for the OFA design, even though the OFA assembly weighs less than the STD assembly. The 14x14 OFA bottom nozzle assembly design is essentially the same as the STD assembly. The amount of flow cross-sectional area is unchanged. The OFA bottom nozzle design has a reconstitutable feature which allows it to be removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the STD nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of the thimble screws as the bottom nozzle is reattached.

As stated in the 17x17 OFA Reference Core Report, Reference 2, for a given burnup, the magnitude of rod bow for the Westinghouse OFA is conservatively assumed to be the same as that of a Westinghouse STD fuel assembly. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation. For the OFA reduced grid forces, a larger fuel tube thickness-to-diameter ratio (t/d) and wider channel spacings between fuel rods as compared to the STD should tend to decrease rod bow occurrences.

The wear of fuel rod cladding is dependent on both the support provided by the assembly skeleton and the flow environment to which it is subjected. Hydraulic flow tests as described in Section 5.0 were performed to verify the compatibility of the 14x14 OFA and STD fuel assembly designs. The results of the tests showed that no significant fuel rod clad wear occur on either the OFA or the STD fuel rod, due to the small amount of crossflow between fuel assemblies.

4.0 NUCLEAR EVALUATION

The key safety parameters evaluated for the conceptual transition and full OFA designs show that the expected ranges of variation for many of the parameters will lie within the normal cycle-to-cycle variations observed for past STD fuel reload designs. The parameters which fall outside of these ranges are those which are sensitive to fuel type, e.g., the moderator temperature coefficient. The accident evaluation, documented in Section 6, has considered ranges of parameters which are appropriate for the transition cycles and beyond.

Power distributions and peaking factors are primarily loading pattern dependent. The usual methods of enrichment variation, spent burnable poison usage, and fresh burnable poison usage can be employed in the transition and full OFA cores to ensure compliance with the peaking factor Technical Specifications.

The methods and core models used in the reload transition analysis are identical to those employed and described in References 1, 2, and 4. These are the same methods and models which have been used in past Point Beach reload cycle designs. No changes to the nuclear design philosophy, methods, or models are necessary because of the transition to OFA fuel.

Future Point Beach reload designs will employ the Relaxed Axial Offset Control Strategy (RAOC) instead of the (Constant Axial Offset Control) CAOC strategy. The RAOC evaluation methodology employed is identical to that generically approved in Reference 6.

A number of changes to the plant Technical Specifications will be implemented as part of the transition to OFA fuel. Some of these changes, whether directly related to OFA fuel or not, impact the core nuclear design. These changes included: (1) positive moderator temperature coefficient, (2) the 0.3 multiplier in the $F_{\Delta H}$ limit function, (3) the change in control rod insertion limits, and (4) implementation of the RAOC strategy.

These Technical Specification changes were considered in the core safety evaluation and it was concluded that the transition and all OFA cores do not result in the current FSAR limits for any accident to be exceeded.

5.0 THERMAL AND HYDRAULIC EVALUATION

5.1 HYDRAULIC COMPATIBILITY

The hydraulic effects of having a mixed OFA and STD fueled core were evaluated by performing hydraulic tests at the Westinghouse Fuel Assembly Test System (FATS) facility. Flow tests on 14x14 OFA, 14x14 STD, and side-by-side OFA and STD fuel assembly arrangements were performed at conditions which approximated reactor conditions. Test results provided information on pressure drops, fuel vibrations and fuel rod clad wear. Based on these test results, it was concluded that hydraulic compatibility existed between the OFA and STD fuel assemblies.

5.2 CALCULATIONAL METHODS

Standard Fuel

Calculational methods currently used on the 14x14 STD fuel assemblies as described in the FSAR and the fuel densification documents remain applicable to the 14x14 STD fuel assemblies in a core containing both 14x14 STD and 14x14 OFA. These include the use of the L-Grid DNB correlation and a design limit DNBR of 1.30. The analysis includes the application of the "old" densification model, consisting of a power spike and pellet eccentricity penalties, the use of pitch reduction, use of a conservative value for the thermal diffusion coefficient, and a margin to the L-Grid DNB correlation limit of 1.24. This results in a generic margin of 18.1%, which is more than sufficient to accommodate rod bow and mixed core DNB effects.

Optimized Fuel

The analysis of OFA considers an all OFA core and includes the following differences from the methods used to analyze the STD fuel: the use of the WRB-1 DNB correlation; the "Improved Thermal Design Procedure," and the THINC IV computer code.

The WRB-1 Correlation provides a significant improvement in Critical Heat Flux (CHF) predictions over previous DNB correlations (References 7 and 8). The 17x17 OFA DNB tests showed that the WRB-1 Correlation correctly accounted for the geometry changes in going from the 17x17 0.374" rod O.D. design to the 17x17 0.360" rod O.D. design, and that the design limit of 1.17 was still applicable, Reference 8. The 14x14 OFA design involved very similar geometry changes from the 14x14 STD fuel design, namely, the reduction of the rod O.D. from 0.422" to 0.400" and the incorporation of a grid design with an increased height and strap thickness due to the change from Inconel to Zircaloy. Confirmatory DNB tests done on the 14x14 OFA typical cell geometry verified that the WRB-1 Correlation accurately predicted CHF values for this geometry type and that the design limit of 1.17 was still appropriate.

The design method employed to meet the DNB design basis is the "Improved Thermal Design Procedure," Reference 9. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability that the minimum DNBR will be greater than or equal to 1.17 for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. For this application, the minimum required design DNBR values are 1.32 for thimble coldwall cells (three fuel rods and a thimble tube) and 1.33 for typical cell (four fuel rods) without consideration of rod bow and transition core effects.

In addition to the above considerations, a specific plant DNBR margin has been considered in the OFA analysis. In particular, the minimum DNBR values of 1.65 and 1.66, for thimble and typical cells respectively, are employed in the safety analyses. The DNBR margin between the minimum safety analysis DNBR values (1.65 and 1.66) and the design DNBR values (1.32 for thimble cells and 1.33 for typical cells) will be used for the flexibility in the design, rod bow DNB effects, and transition core DNB effects of the OFA fuel.

The DNBR margin is defined as

$$\text{Safety analysis DNBR value} = \frac{\text{Design DNBR value}}{1 - \text{Margin}}$$

The THINC IV computer program was used to perform thermal and hydraulic calculations. THINC IV calculates coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions. The THINC IV code is described in detail in References 13 and 14, including models and correlations used. In addition, a discussion on experimental verification of THINC IV is given in Reference 14.

5.3 ROD BOW

The DNB analysis of the STD fuel for the Point Beach Units currently has more than enough generic DNB margin to accommodate rod bow DNB penalties. For comparison purposes, rod bow for the OFA can be predicted based on a comparison of 1/I (I = fuel rod moment of inertia) and the initial rod-to-rod gap for each assembly type. This comparison indicates that the fractional closure, at any given burnup, would be essentially the same in both cases. The 1/I ratio is higher for the OFA, but the initial rod-to-rod gap is also larger. Thus, for a given burnup, the rod bow penalty that is applied to the OFA is no greater than that applied to the 14 x 14 STD fuel. Therefore, as indicated in Section 5.2, both the STD and OFA safety analyses contain sufficient DNBR margin to offset rod bow penalties and transition core effects.

5.4 TRANSITION CORE DNB METHODOLOGY

The 14x14 OFA has a larger hydraulic diameter and flow area compared to the 14x14 STD fuel assembly. Thus, if it is assumed that the same mass flow exists in a STD fuel assembly and an OFA, and there is no allowance for flow redistribution to occur, the STD fuel assembly will have a higher velocity in the rod bundle. The higher velocity, together with the lower value of rod bundle hydraulic diameter, will cause the rod bundle pressure drop to be higher in the STD fuel assembly. Thus, for the same value of mass flow rate into an adjacent set of STD and OFA, the flow would have a tendency to redistribute from the STD to the OFA in the rod bundle region.

In the gridded regions, however, the OFA has a higher value of mixing vane grid loss coefficient. This will induce localized flow redistribution from the 14x14 OFA to the 14x14 STD at the axial zones near the mixing vane grid positions.

The net consequence of this flow redistribution on DNBR is primarily due to the effect this redistribution has on the hot channel mass velocity and the local quality. Depending on the axial location of the minimum DNBR, a DNB penalty can be postulated on either type of fuel assembly when compared to a full core of similar fuel.

A 1 percent transition core DNB penalty was determined to be applicable to both fuel types by analyzing different assembly loading patterns at various core conditions in a manner consistent with previously approved analysis, Reference 15.

Thus the transition cores will be analyzed in the following manner: the STD fuel in a transition core will be analyzed as a full core of STD fuel applying a one percent DNB transition core penalty; and the OFA fuel in a transition core will be analyzed as full core of OFA fuel applying a one percent DNB transition core penalty.

The DNB margins previously described for the STD and OFA fuel are more than enough to accommodate the transition core penalty and rod bow penalty.

5.5 FUEL TEMPERATURES

The fuel temperatures for use in the safety analysis for the 14x14 OFA are not significantly different from those for the 14x14 STD fuel. Small differences in fuel average temperature and peak fuel centerline temperature will not adversely effect the safety analysis calculations. Westinghouse uses the PAD fuel performance code, Reference 11, to perform both design and licensing calculations. When the PAD code is used to calculate fuel temperatures that are to be initial conditions in the safety analysis, a conservative thermal safety model is used.

6.0 ACCIDENT EVALUATION

6.1 INTRODUCTION AND SUMMARY

This section addresses the impact on accident analyses of the following proposed changes for Point Beach Units 1 and 2:

- Optimized Fuel
- Positive Moderator Temperature Coefficient
- $F_{\Delta H}$ Multiplier Change
- Relaxed Axial Offset Control
- Rod Drop Time Increase
- Refueling Shutdown Margin Decrease

Analyses presented in Sections 6.2 and 6.3 are consistent with the methodology presented in Reference 2 for the OFA design. The accident analyses for the OFA design were shown to provide acceptable results by meeting the applicable criteria, such as, minimum DNBR, peak pressure, and peak clad temperature.

Optimized Fuel

For the OFA assembly, the principal mechanical design characteristic which could have an effect on accidents is the smaller fuel rod. This leads to a higher fuel rod temperature, a higher surface heat flux, and a DNB penalty. The larger hydraulic diameter and lower coolant flow velocity cause a reduction in heat transfer after DNB. The smaller fuel rod also leads to a faster heatup rate for severe reactivity transients, such as RCCA ejection.

As a result of the smaller fuel rod, for the same power level, the OFA assembly will have a lower DNB ratio than the STD assembly.

The DNB penalty was offset for the OFA core through the use of the WRB-1 DNB correlation, Reference 7, and the Improved Thermal Design Procedure,

TABLE 6-1

FSAR Chapter 14 ACCIDENT ANALYSIS
SENSITIVITY TO PROPOSED CHANGES

Accidents	OFA	+MTC	Scram Time	Refueling SDM
1. Uncontrolled Rod Withdrawal from a Subcritical Condition FSAR Section 14.1.1.	X	X	X	
2. Uncontrolled RCCA Withdrawal at Power. FSAR Section 14.1.2.	X	X		
3. Rod Cluster Control Assembly (RCCA) Drop FSAR Section 14.1.3.	X			
4. Chemical and Volume Control System Malfunction FSAR Section 14.1.4.	X	X		X
5. Startup of an Inactive Reactor Coolant Loop FSAR Section 14.1.5.				
6. Reduction in Feedwater Enthalpy Incident FSAR Section 14.1.6.	X			
7. Excessive Load Increase Incident FSAR Section 14.1.7.	X	X		

TABLE 6-1 (Con't)

FSAR Chapter 14 ACCIDENT ANALYSIS
SENSITIVITY TO PROPOSED CHANGES

Accidents	OFA	+MTC	Scram Time	Refueling SDM
8. Loss of Reactor Coolant Flow FSAR Section 14.1.8.	X	X	X	
9. Loss of External Electrical Load FSAR Section 14.1.9.	X	X		
10. Loss of Normal Feed-water FSAR Section 14.1.10.				
11. Loss of All AC Power to the Station Aux-iliaries FSAR Section 14.1.11.				
12. Rupture of a Steam Pipe FSAR Section 14.2.5.	X			
13. Rupture of a Control Rod Mechanism Housing-RCCA Ejection FSAR Section 14.2.6.	X	X	X	
14. LOCA FSAR Section 14.3.1	X			

Reference 9. Those transients impacted by OFA fuel are shown in Table 6-1. Further discussion of LOCA can be found in Section 6.3.

Positive Moderator Temperature Coefficient

The present Point Beach Technical Specifications require the moderator temperature coefficient (MTC) to be zero or negative at all times while the reactor is critical. This requirement is overly restrictive, since a small positive coefficient at reduced power levels could result in a significant increase in fuel cycle flexibility, but would have only a minor effect on the safety analysis of the accident events presented in the FSAR.

The proposed Technical Specifications change, noted in Section 7, allows a +5 pcm/°F MTC below 70 percent of rated power, changing to a 0 pcm/°F MTC at 70 percent power and above. A power-level dependent MTC was chosen to minimize the effect of the specification on postulated accidents at high power levels. Moreover, as the power level is raised, the average core water temperature becomes higher as allowed by the programmed average temperature for the plant, tending to make the MTC more negative. Also, the boron concentration can be reduced as xenon builds into the core. Thus, there is less need to allow a positive coefficient as full power is approached. As fuel burnup is achieved, boron is further reduced and the MTC will become negative over the entire operating power range.

The impact of a positive MTC on the accident analyses presented in Chapter 14 of the Point Beach FSAR, Reference 5, has been assessed. Those incidents which were found to be sensitive to minimum or near-zero moderator coefficients were reanalyzed. In general, these incidents are limited to transients which cause reactor coolant temperature to increase. With one exception, the analyses presented herein were based on a +5 pcm/°F MTC, which was assumed to remain constant for variations in temperature.

The control rod ejection analysis was based on a coefficient which was at least +5 pcm/°F at zero power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code, on which the analysis is based, is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature. For all accidents which were reanalyzed, the assumption of a positive moderator temperature coefficient existing at full power is conservative, since as noted in Section 7.0, the proposed Technical Specification change requires that the coefficient be zero or negative at or above 70 percent power.

Accidents not reanalyzed included those resulting in excessive heat removal from the reactor coolant system for which a large negative moderator coefficient is conservative, and those for which heatup effects following reactor trip are investigated, which are not sensitive to the moderator coefficient. Those transients impacted by OFA fuel are shown in Table 6-1.

A recently approved licensing application for a change to positive MTC is given in Reference 12.

FΔH Multiplier Change

A proposed change from K=0.2 to K=0.3 in the following equation for the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) was evaluated with regard to its effect on accident analyses:

$$F^N_{\Delta H} \leq 1.58\ [1.0 + K(1-P)]$$

Where P = the fraction of full power and the K multiplier is the power correction constant.

The effect on accident analyses is determined by the core safety limit changes occurring at very high pressure and low power levels. Since the steam generator safety valves prevent the plant from reaching these limiting conditions, the protection setpoints are unaffected by this change. The change sometimes impacts the axial offset envelope such that the f(ΔI) changes. However, no credit for the f(ΔI) protection is assumed in the accident analyses. Therefore, the safety analyses are not impacted by the proposed FΔH multiplier change.

A recently approved licensing application for a change in the $F_{\Delta H}$ multiplier is given in Reference 12.

Relaxed Axial Offset Control (RAOC)

Relaxed Axial Offset Control (RAOC) has been developed to replace Constant Axial Offset Control (CAOC). It provides enhanced operability at reduced power levels without noticeable degradation of safety margins. The impact of RAOC versus CAOC on the following safety limits has been examined: Maximum KW/FT and minimum DNBRs. The LOCA impacts the RAOC limits rather than LOCA being impacted by RAOC. The non-LOCA transients are impacted through the OTΔT and OPΔT flux-imbalance penalty (f(ΔI)). For those DNB transients that do not trip on the ΔT trips, the minimum DNBRs are impacted by the axial power distribution assumed in the DNBR evaluations. However, in no circumstances does RAOC violate the thermal design basis and no credit for the f(ΔI) protection is assumed in accident analyses. There may be a small degradation in shutdown margin; however, Point Beach has sufficient shutdown margin available to absorb this decrease. Initial assumptions used in the FSAR Chapter 14 Accident Analysis do not change as a result of the proposed RAOC spec. Therefore, there is no impact on accident analyses.

Rod Drop Time Increase

Transients that will be most affected by an increased scram time are the "fast" transients for which the protection system responds by tripping the reactor within a few seconds after the transient begins.

The safety evaluations address the impact of the increased dropped rod time for Point Beach Units 1 and 2. Accidents sensitive to the increase were reanalyzed, and all accidents reanalyzed, whether impacted or not, assumed the increased drop time.

Reduction in Refueling Shutdown Margin

The proposed Technical Specification change for Refueling Shutdown Margin from 10 to 5 percent has been determined to be sufficient for most Westinghouse plants. From a safety perspective, the 5 percent shutdown margin has been shown to be acceptable for the Boron Dilution accident.

Additional Considerations

The accident analyses/evaluations were performed to encompass both Point Beach Units 1 and 2. This includes the following considerations:

- Types of Core
  - Full Standard Core
  - Transition Core
  - Full OFA Core

- Operating Pressures
  - Nominal Pressure of 2250 PSIA
  - Reduced Pressure of 2000 PSIA

- Steam Generators
  - Unit 1 - Model 44F
  - Unit 2 - Model 44

Types of Core

There currently exists a set of applicable safety analyses for the STD fuel core. For those accidents impacted by OFA, reanalysis was performed. Conservatisms in the nuclear design and the thermal-hydraulic parameters used in the accident analysis for the STD and OFA cores bound the transition core. Penalities included in the STD core accident analysis result in values which are within the current limits except for the moderator temperature coefficient and the reactivity insertion time. For those transients sensitive to positive moderator temperature coefficient and scram times, the analyses performed for an OFA core bounded the STD core.

Operating Pressures

The two operating pressures of 2250 psia and 2000 psia were addressed through assumptions used in the reanalysis. For DNB limiting transients, the initial operating pressure of 2000 psia was used. The initial steady state DNBR is lower at the lower pressures and results in a lower transient DNBR. For overpressure transients, an initial pressure of 2250 psia was used yielding a higher peak pressure than that resulting from 2000 psia operation. Analyses currently exist for operation at either pressure. Those accidents not impacted by the proposed changes are still applicable for operation at either pressure as allowed by the Point Beach Technical Specifications.

Steam Generators

The analysis in this report assumes that the transition to OFA fuel in Unit 1 occurs after the replacement of its steam generators. The Steam Generator Repair Report, Reference 10, addresses this impact on accident analyses. The accidents reanalyzed in this report have included effective* tube plugging limits for Units 1 and 2 of 11% and 14% respectively.

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*The combination of sleeved tubes and plugged tubes (such that the effect of the number of sleeved tubes is considered equivalent to a number of plugged tubes) is defined as effective tube plugging.

6.2 NON-LOCA ACCIDENTS

The impact of the proposed changes as identified in Section 5.1 has been assessed for the non-LOCA accidents as provided in Chapter 14 of the Point Beach FSAR. A summary of the impact of each accident and evaluation of the accidents is provided below.

Uncontrolled RCCA Bank Withdrawal From a Subcritical Condition

A control rod assembly bank withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). This incident has been determined to be sensitive to OFA, positive MTC, and rod drop time.

Reanalysis has been performed to determine the impact of the proposed changes. The reanalysis included an upgraded methodology reflecting a more sophisticated and refined calculational model.

Uncontrolled RCCA Bank Withdrawal at Power

An uncontrolled control rod assembly bank withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature (Section 14.1.2 of FSAR). This transient has been determined to be sensitive to OFA and positive MTC. The transient was reanalyzed using the current model of the same digital computer code used in previous Point Beach analysis of this transient.

Rod Cluster Control Assembly (RCCA) Drop

The drop of a control rod assembly results in a step decrease in reactivity which produces a similar reduction in core power, thus reducing the coolant average temperature. Because this transient does not result in a trip, scram time increase has no impact. There is no impact due to positive MTC as a positive MTC would result in a larger reduction in core power level following the RCCA drop. However as a DNB transient, RCCA Drop has been reanalyzed.

Chemical and Volume Control System Malfunction

As stated in Section 14.1.4 of the FSAR, if a boron dilution occurs during refueling or startup, the FSAR shows that the operator has sufficient time to identify the problem and terminate the dilution before the reactor loses shutdown margin. Therefore, since a return to criticality is prevented, the value of the moderator coefficient and scram time has no effect on a boron dilution incident during refueling or startup. Due to the temperature increase at power, a positive MTC would add additional reactivity and increase the severity of the transient. The accident has been reanalyzed to reflect the new refueling shutdown margin of 5% $\Delta k/K$ and a minimum boron concentration for the refueling and startup condition of 1800 ppm. The dilution at power has also been reanalyzed to determine any impact of OFA and positive MTC..

Startup of an Inactive Reactor Coolant Loop

Startup of an idle reactor coolant pump results in the injection of relatively cold water into the core. This accident need not be addressed due to Technical Specification restrictions which prohibit power operation with a loop out of service.

Reduction in Feedwater Enthalpy Incident

The addition of excessive feedwater and inadvertent opening of the feedwater bypass valve result in excessive heat removal incidents which result in a power increase due to moderator feedback. The impact by any proposed changes has been determined to be bounded by the excessive load increase incident.

Excessive Load Increase Incident

An excessive load increase incident is defined as a rapid increase in steam generator flow that causes a power mismatch between the reactor core power and the steam generator load demand. The transient is insensitive to scram time; however, as a DNB transient impacted by OFA, the accident has been reanalyzed. In addition, the BOL cases reflect a positive MTC.

Loss of Reactor Coolant Flow/Locked Rotor

The Loss of Reactor Coolant Flow/Locked Rotor transients have been reanalyzed to determine the impact of OFA, positive MTC, and increased scram time. The positive MTC affects the nuclear power transient, peak reactor coolant system heat flux, pressure, and fuel temperatures. The increased scram time may result in a higher heat flux due to the slower response of the protection system. Combined with the impact of OFA due to a higher fuel rod temperature and surface heat flux, a DNB penalty results.

Loss of External Electrical Load

The result of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in reactor coolant temperature. The consequences of the reactivity addition due to a positive MTC are increases in both peak nuclear power and pressurizer pressure. The OFA impacts the transient as a DNB penalty. Therefore, the Loss of Load has been reanalyzed. This transient is insensitive to rod drop time.

Loss of Normal Feedwater/Station Blackout

This transient is analyzed to determine that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These two events are described together in this section rather than separately, since the manner in which the events are analyzed produces the same transient in both cases.

A loss of normal feedwater may result from feedwater pump failures, valve malfunctions, or from a loss of offsite AC power. The net result of any of these events is a reduction in the capability of the secondary system to remove the heat generated in the reactor core. The reactor is protected from this event by a reactor trip on low feedwater flow or low-low water level in any steam generator, or by an immediate reactor trip if there is a loss of offsite AC power. An auxiliary feedwater system is provided to supply sufficient flow for decay heat removal after reactor trip.

The loss of feedwater due to the loss of offsite AC power differs from a loss of feedwater caused by feedwater pump failures or valve malfunctions in that the loss of AC power will also result in a loss of reactor coolant pump flow. However, the short term effects of this transient during the pump flow coastdown period are already covered by the Complete Loss of Flow incident. The long term effects are taken into account by conservatively assuming an immediate loss of forced reactor coolant pump flow in the analysis of the Loss of Normal Feedwater incident. If there were no loss of AC power for this incident, the coolant flow would remain at its normal value and the reactor would trip via the low-low steam generator level trip with no reduction in DNB ratio below its initial value at the start of the transient.

These events are not sensitive to a positive MTC, since the reactor trip occurs at the beginning of the transient before the reactor coolant system temperature increases significantly. Therefore, these events were not reanalyzed.

Rupture of a Steam Pipe

Since the rupture of a steam pipe is a temperature reduction transient, minimum core shutdown margin is associated with a strong negative moderator temperature coefficient. Therefore, there is no impact by the proposed positive MTC and increase in rod drop time. However, this accident has been reanalyzed to determine the impact on the OFA core with respect to DNB and any impact due to RAOC on shutdown margin.

Rupture of a Control Rod Mechanism Housing-RCCA Ejection

The rod ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions. Since the moderator temperature coefficient is negative at end of life, only the beginning of life cases are impacted by a positive MTC. As a fast reactivity transient, the increase on rod drop time impacts all cases. The impact of OFA is primarily due to the smaller fuel rod which leads to a faster heatup rate. The impact of each of the proposed changes has been addressed by reanalysis.

6.3 LOCA ACCIDENTS

Description of Large Break LOCA Analysis/Assumptions for 14x14 OFA Fuel

The large break loss-of-coolant accident (LOCA) for the Point Beach Plants applicable for transition and full 14x14 OFA core cycles was re-analyzed due to differences in the OFA fuel design. The currently approved 1981 Large Break ECCS Evaluation Model was utilized for this analysis, and three cold leg breaks were re-analyzed.

The LOCA analyses performed assume a full core of 14x14 OFA fuel which conservatively applies to transition cores. The 14x14 OFA is very similar hydraulically to the STD 14x14 fuel it replaces. Differences in total hydraulic resistance between the two designs is approximately 1%. Evaluation of hydraulic mismatches of less than 10% have shown an insignificant effect on blowdown cooling, and current analysis methodology requires conservative increases of hydraulic resistances of 10% or more.

Since the overall resistance between the two types of fuel is so small, only the crossflows due to the smaller rod size and grid designs need be evaluated. The maximum flow reduction due to crossflow calculated to occur in the OFA assembly is ~1.7%. Analyses have been performed which demonstrate that a 5% reduction leads to a maximum PCT increase of 19°F. Therefore, the PCT impact due to crossflow between adjacent 14x14 OFA and 14x14 STD fuel assemblies would be approximately 7°F. This is a small effect and can currently be absorbed in the margin to 2200°F.

The assumption of modeling a full core of OFA is conservative for transition cycles for two major reasons:

1. The increase in core flow area associated with OFA due to the smaller rod diameter has an important impact on flooding rates during reflood. Full OFA core representation decreases core flooding rates, which reduce heat transfer coefficients and result in earlier steam cooling.

2. The OFA fuel assemblies have a higher volumetric heat generation rate than STD fuel. The analysis assumes that an OFA assembly has the hottest rod and maximum $F_{\Delta H}$ which maximizes the calculated PCT.

The analysis techniques used for this evaluation allow for the Point Beach Units to operate at an RCS primary pressure of either 2250 or 2000 psia. A plant specific analysis performed at both pressures demonstrated the 2000 psia case to be limiting; therefore, the results shown here assume an RCS pressure of 2000 psia.

Based on the effect of upper plenum injection for Westinghouse designed 2 loop plants, a 110°F increase in peak clad temperature results from assuming 14x14 OFA fuel for the Point Beach Plant. The methodology employed to develop this penalty was identical to that performed for previous LOCA analyses with STD fuel. This penalty is greater than that calculated for the STD 14x14 fuel design. However, the addition of this PCT penalty on the maximum calculated PCT is well below the 10 CFR 50.46 limit of 2200°F.

The analysis also assumed 14% tube plugging on model 44 steam generators. This analysis result applies to, and bounds, model 44F steam generators up to 11% tube plugging operation.

Results/Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

The Large Break OFA LOCA analysis for Point Beach utilizing the currently approved 1981 Evaluation Models resulted in a PCT of 1938°F at 2.21 F_Q for the 0.4 C_D LOCA. Addition of the UPI injection penalty of 110°F results in a final PCT of 2048°F. The small impact of crossflow for transition core cycles is conservatively evaluated as a 7°F effect, which is easily accommodated in the margin to 10 CFR 50.46 limits.

7.0 TECHNICAL SPECIFICATION CHANGES

In conjunction with the transition to OFA, several changes to the Technical Specifications were proposed. These Technical Specification Changes were submitted previously under separate cover, "Dockets Nos 50-266 and 50-301 Technical Specification Change Request No. 87, Specification for Utilization of Optimized Fuel Assembly Design, Point Beach Nuclear Plant, Units 1 and 2." Letter from C. W. Fay to H. R. Denton, dated March 14, 1983.

The safety evaluations documented in this report support the above Technical Specification Changes.

8.0 REFERENCES

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ATTACHMENT C

STORAGE OF WESTINGHOUSE 14X14 OPTIMIZED FUEL ASSEMBLIES AT THE POINT BEACH NUCLEAR PLANT

I. INTRODUCTION

In preparation for utilization of the Westinghouse 14X14 optimized fuel assembly (OFA) design at Point Beach related storage considerations were reviewed. Because the mechanical and dimensional characteristics of the OFA fuel are compatible with those of the standard design fuel, no complications with OFA fuel handling are anticipated. OFA fuel assemblies, and their inserts will be handled with the same handling tools which are currently in use and available on-site at Point Beach.

The nuclear and thermal-hydraulic characteristics of OFA fuel are, however, somewhat different and therefore it was necessary to verify that storage of OFA fuel assemblies will not adversely affect plant safety.

Specifically the following areas of concern were evaluated:

1. Criticality of new and spent OFA fuel in the Point Beach spent fuel racks and in the new fuel vault;

2. Spent fuel cooling;

3. Radiological effects; and

4. Gamma heating effects on the high density spent fuel racks and spent fuel pool walls.

The safety evaluations for the new fuel vault and the spent fuel pool utilized the same analytical techniques as were used for the licensing of the Point Beach high density spent fuel storage racks. A description of the safety evaluation for the spent fuel pool storage expansion was contained in Technical Specification Change Request No. 34 and related revisions dated March 21, 1978, June 14, 1978, and September 29, 1978.

CRITICALITY CONSIDERATIONS

Spent Fuel Pool

A criticality analysis for the high density fuel racks was performed by Pickard, Lowe, and Garrick, Inc. (PLG) using analytical techniques similar to those used for the licensing of spent fuel racks at other plants, and, in particular, the recent (1979) licensing of the high density Point Beach spent fuel racks for standard fuel. 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 multiplication factor.

The fuel assemblies modeled were presumed to be unirradiated Westinghouse 14 x 14 fuel of OFA design of 4.0 w/o enrichment immersed in water. Taking account of the calculational biases, K_∞ was found to be 0.9058. Adding an additional 0.0042 for tolerances and dimensional uncertainties, the K_∞ under the worst conditions, including calculational uncertainties and biases is 0.9100.

PLG performed the original criticality analyses for licensing the high density spent fuel storage racks in 1979. Their recent criticality analyses for OFA fuel was based upon essentially the same models, computer programs and methodology as was used for the earlier licensing of the high density spent fuel storage racks.

New Fuel Vault

Criticality of OFA in the new fuel vault was evaluated by investigating the neutron multiplication constant as a function of water density under normal storage conditions and under "optimum" moderation conditions.

Under normal storage conditions, the K_∞ of dry storage is extrapolated to be about 0.70. For optimum moderation conditions the peak K_∞ occurs at 3% of full water density. When neutron leakage is accounted for the multiplication factor shifts to 7% of full water density and K_{eff} is then 0.885. Adding in 0.0069Δk for the model bias and 0.0022Δk for calculational uncertainty yields an optimum moderation K_{eff} of 0.894, well under the design limit of 0.98.

For the full flooding condition of the new fuel vault K_{eff} is 0.8630. Adding in the model bias and calculational uncertainty Δk's as before yields a K_{eff} of 0.872.

Thus, although the OFA fuel is somewhat more reactive than the standard design fuel, there is no criticality problem with handling and storage of new and spent OFA fuel for enrichments up to 4.0 w/o. Technical Specification 15.5.4.2 will be changed to reflect a limit of 39.4 gm U-235 per axial centimeter of fuel assembly, consistent with the foregoing analyses.

SPENT FUEL COOLING

A description of the spent fuel pool cooling system was provided with the 1978 licensing submittal for the spent fuel storage expansion. That submittal demonstrated that the spent fuel pool cooling system had the capability of removing the decay heat from the spent fuel pool filled with standard spent fuel under normal and under core unload conditions. Capability to maintain acceptable fuel rod clad temperatures under these conditions was also demonstrated.

To verify the adequacy of the spent fuel cooling system using OFA fuel, calculations were performed using the same methodology as was used for the licensing of standard design spent fuel in the high density spent fuel storage racks. A description of the spent fuel racks currently in use at Point Beach was provided in the earlier licensing submittals. The racks contain a total of 1502 spent fuel assembly storage locations.

The two train spent fuel pool cooling system has a heat removal capability of 15.5×10^6 BTU/hr for one cooling train and 28.2×10^6 BTU/hr with both cooling trains in operation. The two cooling trains have common suction and discharge piping. A description of the Point Beach spent fuel pool heat removal system and its capability presented in the Point Beach FSAR.

Fuel assembly decay heat was calculated in the same manner as was done for the previous submittal based on anticipated OFA fuel cycle plans. The normal refueling case considered the decay heat of 1502 spent fuel assemblies total. The abnormal case included a core unload of 121 assemblies 30 days after the last refueling plus the decay heat from 1381 spent fuel assemblies from normal refuelings.

Because rated plant power and the number of assemblies in the core remains the same, there is little no change anticipated in the fuel assembly decay heat from that originally calculated. The smaller reload region size for the OFA fuel cycle plans may involve a small increase in total spent fuel pool heat load due to the change in the spent fuel discharge schedule. This would be about $.2 \times 10^6$ BTU/hr, much less than the additional 10% conservatism originally assumed. A comparison of the heat loads previously calculated versus the OFA fuel heat loads is as follows:

	Decay Heat Load (BTU/hr)	
	Standard Fuel	OFA Fuel
Normal Refueling (1502 Asemblies)	11.5×10^6	11.7×10^6
Core Unload (1381 + 121 Assemblies)	23.9×10^6	24.1×10^6

One cooling train of the spent fuel cooling system can accommodate the normal refueling heat load. With both trains operating the additional heat load due to a core unload can also be accommodated.

Loss of One Cooling Train

The evaluation of the pool cooling system capability postulates an accident in which one cooling train is assumed to become inoperable shortly after the maximum heat load has been placed in the pool. However, because one cooling train remains operable, heat will continue to be removed from the pool water, and the pool temperature will stabilize at a higher value.

As with standard fuel, an evaluation of the new spent fuel pool heat exchangers has shown that with a pool heat load of 24.1×10^6 BTU/hr, one cooling train can maintain the pool temperature below 150°F. The evaluation assumed a realistic service water inlet temperature of 60°F, a heat transfer coefficient based upon "fouling", and the normal design water flow rates.

Spent Fuel Rack Cooling

The earlier submittal, described the method used to analyze the cooling of standard design fuel. Calculations verifying the ability to adequately cool OFA fuel assemblies in the spent fuel storage racks were performed for the worst case using the same method. The conclusions presented for standard design fuel also apply to use of OFA fuel, namely that:

1. Maximum clad temperature will not exceed the local pool water saturation temperature;

2. Direct gamma heating of the rack, fuel assembly structures and intercell water have been checked and will not cause boiling to occur in the water channel between fuel assemblies; and

3. The spent fuel cooling system provides sufficient cold water so that natural circulation will adequately cool the spent fuel assemblies.

RADIOLOGICAL EVALUATION

Section 7.0 of the spent fuel storage expansion submittal covered the radiological aspects of operation of the Point Beach spent fuel pool with up to 1502 spent fuel assemblies in storage, including a full core unload. The radiological impact of using OFA fuel on the spent fuel storage capabilities has also been evaluated. The isotopic composition of spent fuel was calculated with the ORIGEN computer code using the following conservative assumptions:

1. 50,000 MWD/MTU burnup spread over 4 eleven-month operating cycles and three intervening five-week refueling outages. Burnups were 14,000 MWD/MTU for the first cycle, 16,000 MWD/MTU for the second cycle, 12,000 MWD/MTU for the third cycle, and 8,000 MWD/T for the fourth cycle.

2. Uranium modeling included:

    a. 3.2 w/% initial enrichment
    b. 402,000 grams UO_2
    c. 354,400 grams U
    d. 11,340 grams U-235

3. Weights of structural materials as applicable to an actual Point Beach OFA Plant fuel assembly, including Type 304 stainless steel, Zircaloy-4, carbon steel holddown springs, Zircaloy and Inconel-718 grids.

Maximum pool curie inventories for spent fuel were calculated for completely filled spent fuel storage racks. The maximum configuration is assumed to consist of all but 121 storage locations filled with spent OFA fuel assemblies on the basis of 36 assemblies being discharged every six months. The 121 storage locations are conservatively assumed to be filled with 121 fully burned assemblies from a full core unload. With the conservative assumptions given above, the maximum inventory of radioactivity is 2.03×10^9 curies. Hence, the use of OFA fuel represents a small increase of less than 6% in the maximum inventory of radioactivity.

There is no letdown of the spent fuel pool water and no direct liquid releases. Therefore, use of OFA design fuel will not affect liquid releases. Airborne releases would be slightly affected by the small changes in the spent fuel curie content.

The tritium inventory in OFA will increase. However, the airborne tritium dose to the nearest resident is not expected to change because no significant increase in the total tritium content of the spent fuel pool water is expected. Similarly, the impact of using OFA fuel on Kr-85 releases is negligible. Also the decay rate of the noble gases and radioiodines preclude any significant change in their anticipated release rates. Filtering of particulates will be maintained at about the same level as at present and therefore particulate release rates will not increase as a result of using OFA fuel.

Radiation Levels in the Vicinity of the Spent Fuel Pool

The previous evaluation for standard design fuel resulted in a calculated dose rate of about 1.2 mR/hr with one-year cooling of fuel in the outer row and core unload fuel (three days' cooling) in the immediately adjacent rows. Without a core unload, the maximum dose rates were calculated to be less than 0.3 mR/hr. Because OFA fuel contains less clad and heavy metal absorbing material, these equivalent radiation levels could be somewhat higher. However, the increase is of little significance because standard design fuel assemblies which will be stored in the peripheral storage locations, in accordance with current technical specifications, will effectively reduce radiation levels at the pool wall to values only slightly above those applicable for use of all standard design fuel.

Similarly, the direct radiation dose rate at the surface of the water from the fuel itself would be essentially unchanged because of the large mass of water between the fuel assemblies and the pool water surface. Moreover, the principal source of radiation exposure levels observed at the surface of the pool is due to the concentration of radionuclides in the pool water. Given the same probability of fuel rod leakage, the source of the radiation dose rate is independent of the kind of fuel being stored in the spent fuel pool. The impact on operational dose to workers will be negligible and will be consistent with the ALARA (as low as reasonably achievable) policies implemented at the Point Beach Nuclear Plant.

GAMMA HEATING OF SPENT FUEL STORAGE RACKS

As stated above, direct gamma heating of the spent fuel racks and poison cells was reevaluated to cover storage of irradiated OFA fuel. Since the OFA fuel contains less clad and heavy metal, self-absorption of decay gamma is reduced significantly. This results in greater attenuation and therefore higher temperatures in the materials comprising the spent fuel storage racks.

Calculations indicated the temperature rise of the coolant through the poison box is less than 16°F higher for freshly discharged OFA fuel than for standard fuel using a very conservative model. However, the maximum temperature in the poison slab, the poison box material, and fuel and water box walls will remain less than 213°F, below the local saturation temperature of the water. No boiling will occur and the temperature will remain well below the design limit of 350°F. Note that the Boroflex poison material is structurally stable to 400°F.

Spent Fuel Pool Walls

Earlier analyses for the high density spent fuel storage rack showed that with peripheral fuel assemblies having experienced one year of decay thermal stress in the pool wall would not be significant. Even with an operator error involving placement of a single freshly discharged fuel assembly (three day cooling) on the fuel rack periphery the pool wall thermal stresses would be acceptable. The most limiting condition was placement of two freshly discharged fuel assemblies (3 days decay) adjacent to the poison material sample specimens. It is necessary to place two freshly discharged assemblies adjacent to the poison samples in order to maximize their dose rates consistent with the poison material surveillance program.

With OFA fuel, the shielding effect of the fuel itself will be reduced as discussed above. However, the decay energy of OFA fuel for several months following discharge is significantly less than that of standard fuel. This will offset the effect of reduced fuel self-shielding. Thus, the use of OFA fuel will not cause greater thermal stresses in the pool walls than those caused by standard fuel.

Poison Material Doses

The original calculations were based on two cases; (1) the assumption that spent fuel would not be moved once it was placed in a storage location. Over the lifetime of the plant, this would amount to about three 13-year cycles assuming the spent fuel was removed from the spent fuel storage racks once they became completely filled. (2) The assumption that fuel was moved and a poison slab could see fresh spent fuel placed adjacent to it every six months. This would maximize the doses that the poison slab could experience. In fact, this is the strategy adopted for the poison material surveillance program.

As calculated, the use of OFA fuel would increase the expected dose in the first case from $.18 \times 10^{11}$ Rads to $.20 \times 10^{11}$ Rads. In the second case, the limiting situation, the change to OFA fuel increases the dose from 2.2×10^{11} Rads to 2.5×10^{11} Rads. The original poison surveillance provision was designed to cover exposures of 2.0×20^{11} Rads. This program will be modified to cover exposure of at least 2.5×10^{11} Rads.