Attachment 1

SAFETY EVALUATION

for

INCREASED PEAKING FACTORS AND FUEL UPGRADE

at

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

August 1988

Wisconsin Electric Power Company
Point Beach Nuclear Plant, Units 1 and 2
Docket Nos. 50-266 and 50-301

TABLE OF CONTENTS

LIST OF TABLES

1.0 INTRODUCTION

The Point Beach Nuclear Plant Units 1 and 2 are presently operating with Westinghouse Optimized Fuel Assembly (OFA) and Low-Parasitic (LOPAR) Assembly designs in the core. Westinghouse LOPAR assemblies are also known as standard (STD) fuel assemblies and will be designated as such throughout this submittal.

The Westinghouse 14x14 OFA was reviewed and generically approved by the NRC via Reference 1 and specifically approved by the NRC for Point Beach via Reference 2.

For future fuel cycles, Wisconsin Electric plans to refuel and operate the Point Beach Nuclear Plant Units 1 and 2 with cores containing a modified Westinghouse 14X14 OFA design which incorporates a number of fuel upgrade features and accommodates an increase of the allowable core power peaking factors (F_Q and F-delta-H) specified in the proposed Technical Specifications. This will allow implementation of Low-Low-Leakage Loading Patterns (L4P), a fuel management scheme which will result in a reduction in neutron fluence to the reactor vessel. Fluence reduction will enhance the ability to address current reactor vessel embrittlement issues and to extend the useful life of the Point Beach reactor vessels.

Future operation of the units may involve any combination of the following upgraded fuel product features incorporated into the current Point Beach OFA fuel assembly design:

Removable Top Nozzles (RTNs)

Integral Fuel Burnable Absorbers (IFBAs)

Axial Blankets

Debris Filter Bottom Nozzles (DFBNs)

Extended Burnup Geometry

These upgraded fuel product features are, with the exception of the DFBN, a subset of the VANTAGE 5 design features generically approved by the NRC in Reference 3. The DFBN used for the Point Beach design differs from the VANTAGE 5 Inconel nozzle described in Reference 3, in that it is fabricated from stainless steel and that the size and pattern of the flow holes have been changed. It meets all other design requirements.

Also planned for the Point Beach units is operation incorporating the following reactor core design features:

Low-Low-Neutron-Leakage Loading Patterns (L4P)

Use of Peripheral Power Supression Assemblies (PPSAs)

Removal of fuel assembly thimble plugging devices; and

Elimination of the third line segment of the K(Z) curve.

For convenience, the collection of the above fuel upgrade and core design features will be referred to as "upgraded core features" for the remainder of this report. These upgrade features proposed for the Point Beach units are similar to those approved by the NRC as an amendment to the operating license for the Trojan Nuclear Plant (Reference 4). This report will serve as a reference safety analysis report to support the proposed changes to the Point Beach cores. Sections 3.0 through 7.0 summarize the analyses and evaluations that were performed. Removal of thimble plugging devices from the fuel assemblies and increase of the peaking factors required reanalysis of a number of design basis accidents described in Chapter 14 of the Point Beach Final Safety Analysis Report (FSAR). Results of the reanalysis of the affected accidents and evaluation of the other accidents are described in section 7.0.

The analyses were performed at a core thermal power level of 1518.5 megawatts thermal (MWt) for 2000 psia and 2250 psia operation, with the following additional assumptions made in the analyses/evaluations: a nuclear enthalpy rise hot channel factor (F-delta-H) of 1.70, an increase in the total core peaking factor (F_Q) to 2.50, and removal of the third line segment of the K(Z) curve. The current thermal design flow of 89,000 gpm/loop was used for all analyses except the LOCA analyses, which were conservatively performed for a steam generator tube plugging (SGTP) level of 25% and a corresponding reduction in thermal design flow. Table 1 provides a comparison of major current and proposed design parameters for the Point Beach units.

TABLE 1

COMPARISON OF CURRENT AND PROPOSED POINT BEACH DESIGN PARAMETERS

	Current	Proposed
Fuel type (Westinghouse)	STD, OFA	STD, OFA, upgraded OFA
Core power (MWt)	1518.5	1518.5
Avg. linear power density (kw/ft)	5.7	5.7
System pressure (psia)	2000 (or 2250)	2000 (or 2250)
Core inlet temperature (°F)	545.3 (or 545.0)	545.3 (or 545.0)
Enthalpy rise hot channel		
peaking factor limit (F-delta-H)	1.58	1.70
Total peaking factor limit (F_Q)	2.21	2.50
Total thermal design flow (gpm)	178,000	178,000

2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse standard reload methodology for analyzing cycle-specific reloads (Reference 5), parameters were selected to conservatively bound the values for each subsequent reload cycle and to facilitate determination of the applicability of 10CFR50.59. This report will be used as a basis reference document in support of future Point Beach Reload Safety Evaluations (RSEs) for upgraded core reloads. The objective of subsequent cycle-specific RSEs will be to verify that applicable safety limits are satisfied based upon the reference evaluation/analyses established in this report.

The results of the evaluations/analyses described herein lead to the following conclusions for Point Beach Nuclear Plant Units 1 and 2:

1. Removal of thimble plugging devices from the Westinghouse fuel assemblies containing standard fuel, OFA fuel, or OFA fuel with upgraded fuel product features will satisfy the new design bases and safety limits established by this report. Operation with thimble plugs installed has also been bounded by the analyses.

2. Changes in the thermal-hydraulic and core design characteristics, due to the transition to upgraded fuel product features, will be within the range normally seen from cycle to cycle due to fuel management effects. The change from the current fuel to the upgraded fuel will not cause changes to the current nuclear design bases.

3. The core design, fuel rod design, and safety analyses results documented in this report demonstrate that the core can be operated safely at the current rated design thermal power with an F-delta-H of 1.70, an F_Q of 2.50, a thermal design flow of 39,000 gpm/loop, a reactor coolant pressure of 2000 or 2250 psia, and any combination of the proposed upgraded core features listed in Section 1.0 of this report.

4. The analyses presented herein establish a reference upon which to base reload safety evaluations for future reloads with any combination of the proposed upgraded core features.

3.0 NUCLEAR DESIGN

3.1 Introduction

The nuclear design portion of this submittal has two objectives. First, the impact on the key safety parameters due to the upgraded core features will be evaluated. These safety parameters are used as input to the FSAR Chapter 14 accident analyses. Second, the plant Technical Specifications that apply to nuclear design must be reviewed to determine if they remain applicable or must be revised to accommodate a core containing the upgraded core features.

To satisfy these objectives, a representative core model which contained the upgraded core features was developed using fuel management techniques typical of anticipated Point Beach fuel cycles and the upgraded nuclear design product features previously discussed.

Key safety parameters were evaluated to determine the expected ranges of variation of these parameters and are those described in the standard reload design methodology, Reference 5. The majority of these parameters are insensitive to fuel type and are primarily loading pattern-dependent, e.g., control rod worths and peaking factors. The observed variations in these loading pattern-dependent parameters for the core containing the upgraded core features are typical of the normal cycle-to-cycle variations for core reloads.

3.2 Methodology

The methods and core models used in the Point Beach fuel upgrade analysis are described in References 3, 5, and 6. These licensed methods and models have been used for Point Beach and other previous Westinghouse reload designs. No change to the nuclear design philosophy, methods, or models are necessary because of the upgraded core features. Increased emphasis will be placed on the use of three-dimensional nuclear models because of the axially heterogenous nature of the fuel design when axial blankets and part-length absorbers are used.

The reload design philosophy employed includes the evaluation of the reload core key safety parameters which make up the nuclear design-dependent input to the FSAR safety evaluation for each reload cycle. This philosophy is described in References 3 and 5. These key safety parameters will be evaluated for each Point Beach reload cycle. If one or more of the key parameters fall outside the bounds assumed in the safety analysis, the affected transient will be re-evaluated and the results documented in the RSE for that cycle. The primary objective of the Point Beach upgrade analysis is to determine, prior to the cycle-specific reload design, if the previous key safety parameters will continue to remain applicable. The results of this upgrade core analysis are described in Section 3.3.

3.3 Results

The implementation of L4P, peripheral power suppression assemblies (PPSAs), and axial blankets will impact the core power distributions and peaking factors experienced in the Point Beach cores. The use of axial blankets, where the top and bottom of the enriched fuel stack are replaced by natural uranium pellets and the enrichment of the remaining fuel is increased slightly, results in higher axial peaking factors. The use of L4P and PPSAs results in reduced fluence to the reactor vessel and improved fuel utilization by placing less-reactive fuel on the periphery of the core. The reduction in power carried by the peripheral assemblies is offset by increases in power in the remaining assemblies. The resulting increased radial and axial peaking is accommodated by increasing the core peaking factor limits, F-delta-H and F_Q.

A representative loading pattern was developed and modeled, based on the anticipated upgraded core features. Results of calculations show an increase in radial peaking from previous cycles, which is not unexpected. This results from the reduced power carried by the more highly-burned assemblies placed on the core periphery to reduce neutron leakage, as well as the insertion of blanketed fuel which reduces power at the extreme top and bottom of the fuel, thereby reducing axial leakage.

The total peaking factor, F_Q, was evaluated as a function of core height for the loading pattern consisting of the upgraded core features. Various operating conditions were imposed to achieve variations in power distributions. The limiting values of F_Q times relative power were maintained below the F_Q limit of 2.50 times K(Z), with the assumption that the third line segment of the K(Z) curve is removed. The calculated F_Q values resulted from Relaxed Axial Offset Control (RAOC) analyses performed to determine a revised allowable axial flux difference operating envelope based upon the upgraded core features and the increased F_Q limit. The effects of these changes on Point Beach Technical Specifications are summarized in Attachments 2 and 3. The RAOC analysis methodology is described in detail in Reference 7.

To ensure that the RAOC delta-I band will be conservative for actual upgraded core cycles, a change to the rod insertion limits is also required. The limits will be raised 14 steps (approximately 6 per cent) at all power levels. The change to these limits poses no adverse impact on other safety parameters.

3.4 Conclusion

The key safety parameters evaluated for the conceptual nuclear design show that the expected ranges of variation for many of the parameters will lie within the normal cycle-to-cycle variations observed for reload designs. In addition to the normal variations experienced with different loading patterns, power distributions, and peaking factors show some changes as a result of the incorporation of the upgraded fuel product features and increased peaking factor limits. The usual methods of loading pattern shuffling and enrichment variation can be employed in future cycles using the upgraded core features to ensure compliance with the Point Beach revised Technical Specifications.

The change from the current core to a core containing the upgraded core features will not cause changes to the current nuclear design bases given in the Point Beach FSAR. The evaluation of the Point Beach upgrade demonstrated that the impact of implementing the upgraded core features does not cause a significant change to the physics characteristics of the Point Beach core beyond the normal range of variations seen from cycle to cycle.

4.0 THERMAL AND HYDRAULIC DESIGN

4.1 Introduction

This section describes the thermal-hydraulic analyses performed to support the implementation of upgraded core features in the Point Beach Nuclear Plant, Units 1 and 2. The increase in F-delta-H and the effect of thimble plug deletion was accommodated by using the Departure from Nucleate Boiling Ratio (DNBR) design margin available in the safety analysis DNBR. The thermal-hydraulic design criteria and methods remain the same as those presented in the Point Beach FSAR, with the exceptions noted in the following section. All of the current thermal-hydraulic design criteria are satisfied.

4.2 Methodology

The existing thermal-hydraulic analysis for the Point Beach units is based on the Improved Thermal Design Procedure (ITDP), Reference 8, and the Westinghouse Critical Heat Flux (WRB-1) correlation, Reference 9, as described in the Point Beach FSAR. The analysis of the upgraded fuel product is based on the Revised Thermal Design Procedure (RTDP), Reference 10, and the Westinghouse Critical Heat Flux (WRB-1) correlation. The RTDP removes some of the conservatism in the ITDP methodology, while satisfying the design criterion that protects against DNB in the core. In addition, the W-3 correlation is used where appropriate in both cases.

The DNB thermal design criterion for ITDP or RTDP is that the probability that DNB will not occur on the most limiting fuel rod is at least 95% at a 95% confidence level for any ANSI N18.2 condition I or II event. The Design Limit DNBR is established based on this 95%/95% thermal design criterion. The Design Limit DNBR is then conservatively increased to a Safety Analysis Limit DNBR, which includes a DNBR margin to cover the rod bow penalty as well as future use. The Safety Analysis Limit DNBRs are calculated as follows:

$$\text{Typical (or Thimble) Cell Safety Analysis Limit DNBR} = \frac{\text{Cell Design Limit DNBR}}{1.0 - \text{Margin}}$$

The THINC IV computer program was used to perform thermal and hydraulic calculations, and for calculating 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 11 and 12, including models and correlations used. In addition, a discussion on experimental verification of THINC IV is given in Reference 12

4.3 Hydraulic Compatibility

For thermal-hydraulic purposes, the upgraded fuel product is hydraulically identical to the 14X14 OFA fuel currently used in Point Beach

Units 1 and 2, and no transition core penalty is required. The use of STD fuel assemblies requires a DNBR penalty for conservatism on all the fuel to account for coolant cross flow effects caused by the greater hydraulic resistance of STD fuel. The actual penalty would be very small due to the limited number of STD assemblies to be reinserted.

4.4 Effects of Fuel Rod Bow on DNBR

The phenomenon of fuel rod bowing must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Currently, the rod bow penalty is assessed, based upon References 13, 14, and 15, to be the maximum rod bow penalty for 14x14 OFA. For the desired burnups which are greater than the maximum value discussed in Reference 15, credit is taken for the effect of F-delta-H burndown due to the decrease in fissionable isotopes and the buildup of fission product inventory. Therefore, no additional rod bow penalty is required at the higher burnups.

4.5 DNBR Effect of the Upgraded Fuel

The direct effect on DNBR due to the increase in F-delta-H was offset by the additional margin resulting from RTDP methodology and a revision in the margin which defines the DNBR value for the safety analysis. The new safety analysis DNBR values were selected to retain a margin sufficient to cover rod bow penalty and still provide margin for future use.

The axial blankets and the increased allowable F_Q associated with the upgraded core features affect the axial power distribution and, therefore, the DNBR analyses. These effects were accounted for by means of a limiting axial power distribution in the DNBR analyses for those events which do not trip on the Overtemperature Delta-T (OTDT) reactor trip. The impact on OTDT accident analyses is discussed in Section 7.1.

4.6 Fuel Temperatures for Safety Analysis

The fuel temperatures (as a function of linear heating rate) for use in safety analysis calculations for the upgraded fuel are the same as those used for the current fuel. The PAD fuel performance code, Reference 16, was used for the calculations. The use of IFBAs reduces the fuel temperature, as compared to the use of non-IFBA fuel. This effect is a result of the reduced fuel-to-cladding gap because of the presence of the IFBA coating.

4.7 Thimble Plug Removal

Thimble plugging devices are currently used in the Point Beach units to limit the core bypass flow. All guide thimble tubes that are not in RCCA locations or are not equipped with sources or burnable absorbers currently have thimble plugs inserted in them. A net gain of approximately 2% in DNBR margin is realized due to their presence. When thimble plugs are removed, the design value of core bypass flow increases. This increase was accounted for in the analyses.

Removal of thimble plugs also results in a reduction to the fuel assembly hydraulic loss coefficient. Based on tests performed by Westinghouse, it is estimated that there will be a slight increase in primary system flow rate due to thimble plug removal from the Point Beach cores; however, no mechanical design criteria are impacted by this slight increase in flow rate. Tests also show that there is a net reduction in the hydraulic lift force on a fuel assembly due to a reduced fuel assembly loss coefficient, which more than compensates for the slight increase in vessel flow rate. Thimble plug removal is therefore acceptable from a fuel assembly lift force standpoint.

The effect of thimble plug removal on the core-wide distribution of outlet loss coefficients has been evaluated, and it was demonstrated that the variations in outlet loss coefficient are within the bounds of the sensitivity studies that have been previously performed by Westinghouse. Therefore, thimble plug removal will not result in the reduction of DNBR margin due to mismatches in core outlet pressure gradients and loss coefficients.

This mismatch can also have an effect on fuel rod vibration and wear. Recent Westinghouse fuel rod vibration tests of 17x17 fuel assemblies show that there is no significant difference in fuel rod response between the tests performed with and without the large core outlet loss coefficient mismatch. This can be extended to 14x14 OFA fuel, based upon similarities in lateral flow area/axial flow area ratio, core outlet loss coefficient values (approximately 3.0 for both fuel types), the change in core outlet loss coefficients due to thimble plug removal, and the axial velocity in the fuel rod bundle region. Because of these similarities, it is judged that the core outlet loss coefficient mismatch and maximum crossflow velocities associated with thimble plug removal for any 14x14 OFA will not exceed the test values. Therefore, it is concluded that thimble plug removal will not have a detrimental effect on fuel rod vibration and wear.

An evaluation was performed to determine if thimble plug removal has an adverse effect on the control rods. For the Point Beach upper internals configuration, it was concluded that the maximum core outlet loss coefficient mismatch between an assembly in an RCCA location and an adjacent assembly does not increase with thimble plug removal. Therefore, the magnitude of the crossflow seen by the control rods and the vibration of the rods caused by this crossflow will not be increased. Thus, it is judged that thimble plug removal will not have an adverse impact on control rod wear for the Point Beach units.

In summary, evaluations performed by Westinghouse have shown that the main effect of thimble plug removal is the increase in core bypass flow. This increase has been incorporated into the non-LOCA and small-break LOCA safety analyses, discussed in section 7.0. Based upon the assessment of the impact of thimble plug removal on system and component structural adequacy and core plant safety, it is concluded that, from a thermal-hydraulic standpoint, removal of all or any combination of these devices from the Point Beach cores is acceptable. The evaluation also bounds the use of any combination of dually-compatible thimble plugs, absorber assemblies, peripheral power suppression assemblies, and sources assemblies.

4.8 Conclusion

Thermal and hydraulic analysis has shown that the DNBR penalties resulting from the increase in peaking factor and removal of thimble plugs are offset by the present DNBR margin and the additional margin provided by RTDP methodology. More than sufficient DNBR margin in the safety limit DNBR exists to cover a rod bow penalty and a small transition core penalty. All of the current thermal-hydraulic design criteria are satisfied.

5.0 FUEL ROD DESIGN

5.1 Introduction

The fuel rod design evaluation to support the proposed changes is based on meeting the fuel rod design criteria for the most limiting fuel rod design considered for the Point Beach units. Fuel rod features bounded by these performance evaluations include all combinations of Westinghouse STD and OFA fuel, as currently used in the Point Beach units, and the upgraded fuel product features described earlier.

Increased core power peaking factors affect fuel rod design through increases in the steady-state fuel rod power histories and in the fuel rod transient duty. The fuel rod design criteria affected by this more severe fuel duty are the rod internal pressure, cladding stress and strain, and cladding surface temperature. The evaluation of these design criteria for the bounding Point Beach fuel rod designs and duty shows that the criteria are satisfied for the desired region average burnups.

5.2 Methodology

The fuel rod design criteria are used by Westinghouse to support reliable fuel service for all operations consistent with ANSI N18.2 Condition I and/or Condition II events. The fuel rod design is judged to have met these criteria when it is demonstrated that the performance of a fuel region is within the limits specified by the criteria for these events.

The design criteria are evaluated on a best-estimate-plus-uncertainties basis. Best-estimate results are obtained using NRC-approved best-estimate fuel performance models (References 16 and 17), nominal fuel fabrication attributes, best-estimate powers, fluxes, and fluences. Uncertainties with respect to the design criteria are calculated separately for the significant model, fabrication, and nuclear uncertainties. Typical model uncertainties considered in fuel performance evaluations are fission gas release, helium release, rod growth, cladding creep, fuel densification and swelling, and cladding corrosion. Typical fabrication uncertainties considered are fuel OD, cladding ID and OD, fuel density, plenum size, and backfill pressure. Nuclear design uncertainties in the power, flux, and fluence are also considered. The total uncertainty is obtained by a statistical convolution of the individual uncertainties.

5.3 Results and Conclusion

Evaluations of the rod internal pressure and cladding stress criteria show that these design criteria will be satisfied for the desired increased allowable core power peaking factors and the desired fuel rod design features for the desired region average burnups. Although the cladding surface temperature criteria were shown to be satisfied for fuel rods operated through five annual cycles, the rods are not limited to five annual cycles of operation; reinsertion of assemblies beyond five cycles of operation is addressed on a cycle-by-cycle basis considering the power histories of the rods in question.

6.0 REACTOR PRESSURE VESSEL SYSTEM EVALUATIONS

6.1 Introduction

The evaluations presented in this section were performed to ensure that the use of the modified 14x14 OFA fuel with the removal of thimble plugging devices in the Point Beach units will not violate reactor pressure vessel internals system design requirements.

6.2 Results

Thimble plug removal results in a reduction in core hydraulic resistance and a related increase in the portion of core bypass flow passing through the fuel assembly thimble tubes. These direct consequences lead to secondary effects within the reactor pressure vessel internals system. Such effects were evaluated for fluid system pressure drops, core bypass flow, baffle gap coolant jetting momentum flux, closure head fluid temperature, internals component lift forces, and RCCA drop time.

The evaluations were performed for both system pressures (2000 psia and 2250 psia) using operating, geometric, and hydraulic characteristics specific to Point Beach Units 1 and 2 with Westinghouse 14x14 optimized fuel and the thimble plugging devices removed.

6.3 Conclusions

The impact of thimble plug removal on reactor internal pressure losses, coolant jetting through core baffle plate gaps, and closure head average fluid temperature is essentially inconsequential. Removal of thimble plugs is consistent with the revised total core bypass flow limit of total reactor vessel flow and will result in an insignificant reduction in total reactor internals lift forces.

Thimble plug elimination at Point Beach Units 1 and 2 will not impact the Technical Specification RCCA drop time-to-dashpot-entry limit of 2.2 seconds.

7.0 ACCIDENT ANALYSIS

7.1 Non-Loss-of-Coolant Accidents (Non-LOCAs) Analyses and Evaluations

7.1.1 Introduction

These analyses and evaluations address the impact of the proposed changes discussed in Section 1.0 on non-LOCA events presented in Chapter 14 of the Point Beach Nuclear Plant, Units 1 and 2, FSAR. In addition, they also address use of the Revised Thermal Design Procedure (RTDP) and the new Dropped Rod Methodology (References 10 and 18, respectively). It should be noted that the WCAPs describing the RTDP and the new Dropped Rod Methodology are presently undergoing NRC review, although Safety Evaluation Reports (SERs) are expected to be issued soon.

7.1.2 Effects of Change in F-delta-H

An increase in the power-dependent F-delta-H limit does not directly affect the system transient response of the non-LOCA events presented in the Point Beach FSAR. Rather, the power level-dependent F-delta-H limit is used in the determination of the DNBR for those events for which DNB is the safety acceptance criterion. (The F-delta-H is not relevant for the non-DNB related non-LOCA events.)

DNBR calculations fall into two categories: (1) those events in which the power-level dependent value of F-delta-H is indirectly accounted for via the core DNB safety limits, and (2) those events which directly assume the power level dependent value of F-delta-H in the analysis.

7.1.2.1 Indirect Effect of Change in F-delta-H

For those events in the first category, revised core DNB safety limits were generated reflecting the increased F-delta-H limit of 1.70. Based upon the new core limits, new Overtemperature and Overpower Delta-T (OTDT/OPDT) setpoint equations have been calculated and are reflected in the changes to the Technical Specifications (Attachment 2). The events which rely upon the OTDT/OPDT setpoints for protection have been reanalyzed. These events and corresponding FSAR sections include:

FSAR Section	Event
14.1.2	Uncontrolled RCCA Withdrawal at Power
14.1.6	Reduction in Feedwater Enthalpy Incident
14.1.7	Excessive Load Increase Incident
14.1.9	Loss of External Electrical Load

The results of the analyses of these events show that the calculated DNBR value for each event is greater than the Safety Analysis Limit value. Therefore, the conclusions presented in the FSAR for these events remain valid.

7.1.2.2 Direct Effect of Change in F-delta-H

For those events in the second category, the increased value for F-delta-H was used directly in the analysis of the following events:

FSAR Section	Event
14.1.1	Rod Withdrawal from Subcritical
14.1.3	Dropped Rod
14.1.5	Startup of an Inactive Loop
14.1.8	Loss of Flow

An increase in F-delta-H results in a decrease in the DNBR value for a given set of thermal-hydraulic conditions. However, the results of the analysis of these events, assuming the revised value for F-delta-H, show that the calculated DNBR value for each event is greater than the Safety Analysis Limit value. Therefore, the conclusions presented in the FSAR for these events remain valid.

7.1.2.3 Steamline Break Evaluation

The Rupture of a Steam Pipe event in FSAR 14.2.5 is an ANSI N18.2 Condition IV event. For the Core Response event, it is shown that the DNBR design basis is met. The analysis is performed at zero power conditions, assuming the most reactive rod stuck in its fully withdrawn position. An increase of the power-dependent F-delta-H limit results in an increase in the zero power stuck rod peaking factor. The impact of the increase in the zero power stuck rod peaking factor on the DNBR calculation has been evaluated, and it has been shown that the DNBR design basis has been met. In addition, the increase in F-delta-H will not change the primary-to-secondary heat transfer characteristics in the Mass-and-Energy-Release-to-Containment event. Therefore, this event is not impacted by the increase in F-delta-H, and the conclusions presented in the FSAR remain valid.

7.1.3 Effects of Increase in F_Q

To ensure that cladding integrity and fuel melting at the "hot spot" are maintained within the applicable safety analysis limits, the two affected transients for an increase in the F_Q limit were reanalyzed:

FSAR Section	Event
14.1.8	Locked Rotor
14.2.6	Rod Ejection

The results of the analyses show that all applicable safety criteria are met for both events, and therefore, the increase in F_Q to 2.50 is acceptable with respect to the conclusions presented in the FSAR.

7.1.4 Effects of Thimble Plug Removal

The removal of the thimble plugs will allow coolant flow through the guide thimble tubes, thus reducing the amount of flow available for core heat removal. This is reflected in the increase in the core bypass flow assumed in the safety analyses. The events reanalyzed have incorporated the effects of the increase in core bypass flow. The Steamline Break event was not reanalyzed. However, an increase in core bypass flow and the resultant reduction in core flow would reduce the severity of the core cooldown in this transient. This would result in a lower peak heat flux, which is a benefit with respect to DNB. Therefore, the increase in core bypass flow would not invalidate the conclusions of the Steamline Break Core Response event. In addition, the reduction in core flow would not change the primary-to-secondary heat transfer characteristics of the Mass-and-Energy-Release-to-Containment event. Therefore, thimble plug removal will not impact the Steamline Break Mass-and-Energy-Release-to-Containment event.

The removal of the thimble plugs would also have a slight impact on the vessel pressure drops. The effects of this change have been incorporated into those events which were reanalyzed. For the Steamline Break events, the change in vessel pressure drops would have an insignificant impact on the results of the event. Therefore, the change in the vessel pressure drops would not invalidate the conclusions of the Core Response event, nor would this change impact the Mass-and-Energy-Release-to-Containment event.

7.1.5 Effects of Other Upgraded Core Features

Core flow areas and loss coefficients were preserved in the design of the RTN and DFBN. As such, no parameters important to the non-LOCA safety analyses are impacted, and the conclusions of the non-LOCA safety analyses remain valid.

The effect of axial blankets, IFBAs, and extended burnup on the reload safety analysis parameters is taken into account in the reload design process. The axial power distribution assumption in the safety analyses kinetics calculations have been determined to be applicable for evaluating extended burnup and for the introduction of axial blankets and IFBAs in the Point Beach units.

The use of a low-low-leakage loading pattern and PPSAs will decrease the power at the periphery of the core, resulting in increased peaking factors. The reanalysis of the non-LOCA events has assumed an increase in F-delta-H to 1.70 and an increase in F_Q to 2.50. Since all applicable safety criteria were met with these assumptions, use of loading patterns that adhere to these new design limits is acceptable with respect to non-LOCA safety analyses.

7.1.6 Non-LOCA Safety Evaluation Methodology

The non-LOCA safety evaluation process is described in References 1, 3, and 5. The process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes, on a cycle-by-cycle basis, which may invalidate existing safety analysis assumptions and identifies the transients which require re-evaluation.

Any required re-evaluation identified by the reload methodology is one of two types. If the identified parameter is only slightly out of bounds, or if the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a significantly or not easily quantifiable effect on the transients, reanalyses are required. The reanalysis approach will typically use the analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in FSARs, subsequent submittals to the NRC for a specific plant, reference SARs, or generic report submittals for NRC approval.

The key safety parameters are documented in Reference 5. Values of these safety parameters which bound all three fuel types (STD, OFA, OFA with upgraded features) were assumed in the safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. Re-evaluation of the affected transients would take place and would be documented for the cycle-specific reload design, in accordance with Reference 5.

7.1.7 Conclusions

Using the revised safety analysis assumptions associated with the proposed upgraded core features indicated in section 1.0 of this report, the analyses and evaluations performed show that all applicable safety criteria have been met. Therefore, the conclusions of the non-LOCA safety analyses presented in Chapter 14 of the Point Beach FSAR remain valid.

7.2 Loss-of-Coolant Accident (LOCA) Events

7.2.1 Large-Break Accident

The large-break LOCA event presented in FSAR 14.3.2 is being reanalyzed as part of the Two-Loop Upper Plenum Injection Plant Model development effort (Reference 20). The methodology and the Prairie Island analysis have been submitted to the NRC for approval, which is expected in September 1988. The plant-specific analysis for Point Beach Nuclear Plant, Units 1 and 2, will incorporate the increased peaking factors, thimble plug removal, upgraded fuel product features, and increased steam generator tube plugging. Results of the Point Beach large-break LOCA analysis will be reported separately.

7.2.2 Small-Break Accident

7.2.2.1 Introduction

The small-break LOCA analysis for Point Beach Units 1 and 2 assumed a 4-inch diameter cold leg break. The analysis incorporated the proposed changes discussed in section 1.0 of this report, as well as 25% steam generator tube plugging, a corresponding reduction in thermal design flow, and an elevation-independent F_Q envelope (i.e., flat K(Z) curve).

7.2.2.2 Methodology

The analysis was performed using the NRC-approved Westinghouse NOTRUMP Small Break Evaluation Model (Reference 21) for a 4-inch break size. The Westinghouse NOTRUMP Emergency Core Cooling System (ECCS) Small Break Evaluation Model, developed to determine the Reactor Coolant System (RCS) response to design basis, small break LOCAs, consists of the NOTRUMP and LOCTA-IV computer codes, References 22 and 23, respectively.

The use of NOTRUMP for small-break LOCA analyses of Westinghouse reactors was accepted by the NRC in an SER dated May 21, 1985. Based on previous analyses done for Westinghouse two-loop plants, and NOTRUMP generic studies described in WCAP-11145, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code," submitted to the NRC on June 11, 1986, we believe our reanalysis of a single break size and location is justified. This approach reflects the method being followed by PBNP in the ongoing best-estimate large-break LOCA analysis using W COBRA/TRAC by using Prairie Island, a plant with a similar two-loop Westinghouse NSSS units, as a lead plant. The following reasoning was used to choose the break size and location to be analyzed and to establish the acceptability of the results:

1. Within each evaluation model (EM) used to analyze two-loop small-break LOCAs, the limiting break size and location has always been the same. In the WFLASH 74 EM, the four-inch, cold-leg break was limiting in every case. In the WFLASH Oct. 75 EM, the six-inch, cold-leg break was always limiting. In the NOTRUMP analyses of a generic two-loop case and of Northern States Power's (NSP) Prairie Island plant, the four-inch, cold-leg break was again limiting. Since our analysis was performed using NOTRUMP, we expected a four-inch, cold-leg break to be limiting.

2. The NRC Safety Evaluation Report for the NOTRUMP SBLOCA EM, described in WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," and WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," required that, as part of the submittal for satisfying NUREG-0737 II.K.3.31, confirmation be provided that the limiting break location in NOTRUMP analyses had not shifted away from the cold leg to the hot leg or pump suction leg. Analyses presented in WCAP-11145-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code" showed that the cold leg break location was still limiting for small-break LOCA. We therefore expect the cold-leg break location to be limiting for PBNP.

3. Since the major input parameters used in the analysis of PBNP are similar to those used for Prairie Island (see Table 2), we expect similar results using the same EM. Prairie Island demonstrated a peak cladding temperature (PCT) of 1000°F for the four-inch, cold-leg break and no core uncovery for the three- and six-inch break sizes. As a result, we again expected the four-inch, cold-leg break to be limiting for PBNP.

   To further demonstrate the similarity in results using the NOTRUMP model, a comparison of the inputs used for PBNP, Prairie Island, and the generic two-loop case (see Table 2) shows that although PBNP and Prairie Island are similar in their inputs, the generic case differs in a number of areas. Despite these differences, however, the results of the generic case showed a PCT of 796°F for the four-inch cold leg break, which were the limiting break size and location. This comparison further confirmed our decision to analyze a four-inch cold leg break.

4. We then established a set of criteria to determine the acceptability of our results. First, the results had to indicate some core uncovery in order to have a PCT greater than coolant saturation temperature, otherwise the results would be trivial. The indicated PCT also had to be less than 1600°F. This self-imposed cutoff would ensure significant margin to the 2200°F limit, would prevent any significant zirc-water reaction, and would ensure that the PCT for other break sizes and locations would fall below the 2200°F limit. The latter benefit derives from the historical results of two-loop analyses. In the WFLASH 74 EM, the greatest difference in PCT between break sizes was 600°F. By the WFLASH 75 EM that difference was reduced to 350°F, and in the NOTRUMP analyses that difference became 40°F. Thus a 1600°F cutoff ensures that the 2200°F limit is met, especially for analyses using NOTRUMP.

7.2.2.3 Results

The actual analysis of the four-inch, cold-leg break for PBNP resulted in a PCT of 809°F, well below the established cutoff of 1600°F. A comparison of the PBNP inputs and results with those of the Prairie Island analysis shows that the PBNP PCT is actually lower, due primarily to its lower rated power level. This comparison also indicates that there should be no significant core uncovery for the other break sizes for PBNP. Based on these results, we believe that the plant-specific analysis of a single break -- the four-inch, cold-leg break -- in conjunction with generic two-loop and lead-plant analyses of a spectrum of break sizes and locations adequately demonstrates that the emergency core cooling system satisfies the acceptance criteria of 10 CFR 50.46.

TABLE 2

COMPARISON OF POINT BEACH, PRAIRIE ISLAND, AND GENERIC TWO-LOOP NOTRUMP INPUT ASSUMPTIONS

Point Beach	Prairie Island	Generic Two-Loop
1518.5 MWt Core Power	1650 MWt Core Power	1709.2 MWt Core Power
2000 psi RCS Pressure	2250 psi RCS Pressure	2250 psi RCS Pressure
570°F T_{avg}	570.6°F T_{avg}	573°F T_{avg}
2.50 F_Q	2.50 F_Q	2.32 F_Q
1.70 $F_{\Delta H}$	1.70 $F_{\Delta H}$	1.62 $F_{\Delta H}$
K(Z) Third Line Removed	K(Z) Third Line Removed	K(Z) Third Line Not Removed
25% SGTP*	10% SGTP*	0% SGTP*
Upflow Barrel/Baffle Configuration	Downflow Barrel/Baffle Configuration	Downflow Barrel/Baffle Configuration

*Steam Generator Tube Plugging

The analysis demonstrates that the ECCS satisfies the acceptance criteria of 10CFR50.46 for a 4-inch diameter cold-leg break. That is:

1. The calculated peak fuel element cladding 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 one percent of the total amount of zircaloy in the reactor.

3. The localized cladding oxidation limit of 17 percent 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. This is required to remove the heat from the long-lived radioactivity in the core.

Mixed core hydraulic resistance mismatch is not a significant factor for a small-break LOCA analysis. Therefore, it is not necessary to perform any additional small-break evaluations for transition cores, and it is sufficient to reference the small-break LOCA analysis performed for a core of 14x14 OFA with upgraded core features as bounding all transition cycles.

7.2.3 Steam Generator Tube Rupture Accident

7.2.3.1 Introduction and Methodology

The steam generator tube rupture (SGTR) analysis was performed to evaluate the radiological consequences of an SGTR accident. A complete single tube break adjacent to the steam generator tube sheet was assumed. Since the RCS pressure is greater than the steam generator shell side pressure, radioactive reactor coolant is discharged into the secondary system. For the Point Beach units, the major factors that affect the resultant offsite doses are the amount of fuel defects (level of reactor coolant contamination), the primary-to-secondary mass transfer through the ruptured tube, and the steam released from the ruptured steam generator to the atmosphere.

Since the conservative fuel failure assumption of 1% defective fuel for the Point Beach SGTR analysis will not change due to the proposed changes, the variables which impact the offsite radiation doses calculated for the FSAR SGTR analysis are the primary-to-secondary break flow and the steam released from the ruptured steam generator to the atmosphere.

As a first step in evaluating the impact of the proposed changes on the Point Beach FSAR SGTR results, the FSAR SGTR analysis was conservatively re-evaluated to reflect an update to the safety injection termination requirements in the current Point Beach SGTR recovery procedures. Specifically, it was assumed that full safety injection flow is maintained to the RCS from the time of safety injection initiation until 30 minutes after the tube rupture, when the RCS and ruptured steam pressure are assumed to equilibrate, and break flow is assumed to be terminated.

Subsequently, SGTR sensitivity analyses were performed to assess the impact of the proposed changes on the primary-to-secondary break flow and steam released to the atmosphere via the ruptured steam generator. The results of these analyses were then used to determine the change to the offsite radiation doses reported in the FSAR for the SGTR accident.

7.2.3.2 Results

The Point Beach FSAR SGTR re-evaluation indicates an increase in the primary-to-secondary break flow and steam released via the ruptured steam generator, above those reported in the FSAR, due to the updated safety injection termination assumption. The SGTR sensitivity analyses for the proposed changes at Point Beach show a further, but slight, increase in the primary-to-secondary break flow and steam released via the ruptured steam generator. These increases for the sensitivity analyses are for the combined effect of all changes desired, although the results of the sensitivity analyses indicate that the 25% steam generator tube plugging assumption was the foremost contributor to the increase in break flow and mass release.

These results have been used to calculate the offsite doses to determine the effect of the proposed changes on the offsite radiological consequences reported in the FSAR.

7.2.3.3 Radiological Consequences

A design basis failure of a single steam generator tube was evaluated, and the assumptions made for the radiological analysis are consistent with those used in the FSAR analysis, with the exception of the pre-existing primary-to-secondary leak rate and the corresponding secondary coolant iodine activity. The revised leak rate assumed is consistent with the Point Beach Technical Specifications.

The doses calculated for the SGTR re-evaluation and sensitivity analyses remain within a "small fraction" of the 10CFR100 exposure guidelines, which are 300 rem thyroid and 25 rem whole body. This "small fraction" is defined as 10% of the guideline value, that is, 30 rem thyroid and 2.5 rem whole body, and is the smallest of the exposure limits defined by NRC in NUREG-0800.

7.2.3.4 Conclusion

Based upon the results of the Point Beach SGTR re-evaluation and the sensitivity analyses, the conclusion in the Point Beach FSAR that the SGTR radiological consequences are within a small fraction of the limits set forth in 10CFR100 is still valid.

8. REFERENCES

1. Davidson, S.L., and Iorii J.A., "Reference Core Report - 17x17 Optimized Fuel Assembly," WCAP 9500-A, May, 1982.

2. Letter from J.R. Miller (USNRC) to C.W. ay (WEPCO), subject: Revised Technical Specifications to Allow Use of Westinghouse OFAs in Point Beach Reloads, October 5, 1984.

3. Davidson, S.L., and Kramer, W.L., Ed., "VANTAGE 5 Reference Core Report VANTAGE 5 Fuel Assembly," WCAP 10444-P-A, September 1985.

4. Letter from D.W. Cockfield (PGE) to NRC Document Control Desk, subject: Trojan Nuclear Plant License Change Application #161, November 20, 1987.

5. Davidson, S.L. (Ed.), et.al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

6. Davidson, S.L. (Ed.), et.al., "ANC: Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

7. Miller, R.W., et.al., "Relaxation of Constant Axial Offset Control," WCAP-10216-P-A, August 1982.

8. Chelemar, H. et.al., "Improved Thermal Design Procedure," WCAP-8567, July 1975.

9. Motely, F.E., et.al., "New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids," WCAP-8262-P-A and WCAP-8763-A, July 1984.

10. Friedlanc, A.J., Ray, S., "Revised Thermal Design Procedure," WCAP-11397, February 1987.

11. Hochreiter, L.E., Chelemer, H., Chu, P.T., "THINC IV, An Improved Program For Thermal Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

12. Hochreiter, L.E., "Application of the THINC IV Program to PWR Design," WCAP-8054, October 1973.

13. Skaritka, J., (Ed.), "Fuel Rod Bow Evaluation," WCAP-8691, Revision 1 (Priprietary), July 1979.

14. "Partial Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E.P. Rahe, Jr. (Westinghouse) to J.R. Miller (NRC), NS-EPR-2515, dated October 9, 1981; "Remaining Response to Request Number 1 for Additional Information on WCAP-8691, Revision 1" letter, E.P. Rahe, Jr. (Westinghouse) to J.R. Miller (NRC), NS-EPR-2572, dated March 16, 1982.

15. Letter from C. Berlinger (NRC) to E.P. Rahe, Jr. (Westinghouse), "Request for Reduction in Fuel Assembly Burnup Limit for Calculation of Maximum Rod Bow Penalty," June 18, 1986.

16. Miller, J.V., (Ed.), "Improved Analytical Models Used In Westinghouse Fuel Rod Design Computations," WCAP-8720, October 1976 (Proprietary) and WCAP-8785, October 1976 (Non-Proprietary).

17. Weiner, R.A., et.al., "Improved Fuel Performance Models for Westinghouse Fuel Design and Safety Evaluations," WCAP-10851, June 1984 (Proprietary).

18. Haessler, R.L. et.al., "Methodology for the Analysis of the Dropped Rod Event," WCAP-11394 (Proprietary), WCAP-11395 (Non-Proprietary), April 1987

19. Davidson, S.L., Kramer, W.R., (Ed.), "Extended Burnup Evaluation of Westinghouse Fuel," WCAP-10125-P-A, December 1985.

20. Dederer, S.I., et.al., "Westinghouse Large-Break LOCA Best-Estimate Methodology," WCAP-10924-P (Volumes 1 and 2), April 1988.

21. Lee, N., et.al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A and WCAP-11081-A, August 1985.

22. Meyer, P.E., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A (Priprietary) and WCAP-10080-A (Non-Proprietary), August 1985.

23. Bordelon, F.M., et.al., "LOCA-IV Program: Loss of Coolant Transient Analysis, "WCAP-8301 (Proprietary), and WCAP-8305 (Non-Proprietary), June 1974.

ATTACHMENT 2

TECHNICAL SPECIFICATION CHANGE REQUEST 127

AUGUST 1988

PROPOSED TECHNICAL SPECIFICATION CHANGES