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U. S. NUCLEAR REGULATORY COMMISSION
Document Control Desk
Washington, D. C. 20555

Attention: Mr. George Lear, Project Director
PWR Project Directorate 1

Gentlemen:

DOCKET 50-266
CYCLE 15 RELOAD
POINT BEACH NUCLEAR PLANT, UNIT 1

A refueling shutdown for the Point Beach Nuclear Plant Unit 1 began on April 4, 1987. This shutdown was at the end of Unit 1 Cycle 14 operation. Cycle 14 burn-up was approximately 10,100 MWD/T. Point Beach Unit 1 start-up for Cycle 15 is expected to occur in May 1987 following an eight-week refueling and maintenance outage.

Reload Region 17 for Unit 1 Cycle 15 operation will contain 32 Westinghouse 14 x 14 Optimized Fuel Assemblies (OFA). This will be the third reload region of OFA fuel inserted into the Unit 1 core. The use of OFA fuel in both Point Beach Nuclear Plant units was reviewed and approved, as reported in the NRC Nuclear Safety Evaluation Report issued on October 5, 1984, in support of License Amendment 86 (Technical Specification Change Request 87) for Unit 1.

The Cycle 15 core will contain four 8-rodlet "water displacer" assemblies. The "water displacer" rod assemblies are essentially the same as the burnable absorber tubes (BPRA's), which contain borosilicate glass as a burnable absorber, except that the "water displacer" tubes contain no absorber and are pressurized with helium. Engineering review of the "water displacer" design has verified that, even if leakage occurs, no design or safety concerns are posed.

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The mechanical and thermal-hydraulic design for the Unit 1 Cycle 15 reload core is similar to that of previously reviewed and accepted reload designs containing OFA fuel. This core is designed to operate under nominal design parameters and the approved Technical Specifications, including those provided with License Amendment 86 for Unit 1, so that the core characteristics will be less limiting than those previously reviewed and accepted. For those postulated accidents presented in the FSAR which could be affected by the reload core, re-evaluation has demonstrated that the results of the postulated events are within allowable limits. The reload core meets the $F_Q^T \times P$ limit of less than 2.21 x K(z) and the $F_{\Delta H}^N$ limit of less than 1.58.

In accordance with past practice, the Westinghouse reload safety evaluation report relies on previously reviewed and accepted analyses as reported in the FSAR, in the OFA transition safety reports, and in earlier reload cycle safety evaluation reports. The reload safety evaluation report for Unit 1 Cycle 15 demonstrates that no unreviewed safety questions, as defined in 10 CFR 50.59, are involved in the operation of Unit 1 during Cycle 15. No application for license amendments beyond those already approved by the NRC in License Amendment 86 are, therefore, required for Cycle 15 operation. This 10 CFR 50.59 evaluation will be confirmed by the Manager's Supervisory Staff prior to start-up of Cycle 15. Verification of the core design will be performed by means of the standard start-up physics tests normally conducted at the beginning of each cycle.

The barrel-baffle region upflow conversion project, as described in the attachment, will be completed for Unit 1 during this outage. This conversion is being performed to alleviate baffle-jetting fuel damage in Unit 1, as described in Licensee Event Report 85-002-01. The safety evaluation report for this project demonstrated that no Technical Specification changes or unreviewed safety questions result from this modification. This safety evaluation will be confirmed by the Manager's Supervisory Staff in accordance with 10 CFR 50.59 prior to start-up of Unit 1 Cycle 15.

Please contact us if you have any questions regarding the Cycle 15 reload design or operation.

Very truly yours,

CW Fay

C. W. Fay
Vice President
Nuclear Power

Attachment

Copy to NRC Resident Inspector
NRC Regional Administrator, Region III

ATTACHMENT

POINT BEACH NUCLEAR PLANT-UNIT 1

UPFLOW CONVERSION

Introduction

In April of 1985 a failed fuel rod was discovered in Unit 1 adjacent to a baffle plate joint with full length bolting (Licensee Event Report 85-002). In October of 1985 several failed fuel rods were found in assemblies located adjacent to two rabbeted baffle plate joints in Unit 2 (LER 85-004). Both of these occurrences are attributed to the phenomenon known as baffle jetting. Baffle jetting occurs when a high differential pressure exists across the baffle plates in the lower internals. This high differential pressure causes a jet of water to be directed horizontally into the core region through small gaps which exist in the joints between individual baffle plates. This jet of water can potentially cause vibration-induced damage to the fuel rods.

The upflow modification is intended to eliminate this problem by reducing the differential pressure across the baffle plates, thus reducing coolant jetting below the level that can cause fuel damage. Differential pressure will be reduced by reversing the flow direction in the barrel-baffle region from down to up, as discussed in the next section.

Upflow Conversion

The region between the core barrel and the baffle plates (the "barrel-baffle" region) must be cooled due to heat transfer from the core region and direct gamma heating. This cooling function is provided by a secondary flow path through holes in the core barrel just below the top former plate and holes in the intermediate former plates (Figure 1). Reactor coolant enters the core barrel holes and flows down through the barrel-baffle region to the bottom of the core where it turns and goes up through the core. This flow pattern results in a large differential pressure across the baffle plates due to the pressure drops that occur as water flows down through the barrel-baffle region and up through the core.

The upflow modification will significantly reduce this differential pressure by reversing the flow path through the barrel-baffle region. This modification involves plugging the core barrel holes and machining new holes in the top former plate (Figure 2). In the new flow path for the barrel-baffle region, reactor coolant enters at the bottom of the core barrel (between the baffle plates and the lower core plate), flows upward through the former plate holes, and exits through the new holes in the top former plate joining the main coolant in the outlet plenum (Figure 1). The resulting axial pressure drop through the barrel-baffle region more closely matches the axial pressure drop through the core, thus reducing the net transverse pressure drop across the baffle plates.

Evaluation

Westinghouse performed various analyses of the effects of the upflow modification on PBNP. The results of these structural, thermal-hydraulic, and accident analyses are outlined in Westinghouse's Upflow Licensing Safety Evaluation Report for Point Beach Nuclear Units 1 and 2, September 1986. These results are considered satisfactory with respect to the design basis of the Point Beach Nuclear Plant. Westinghouse demonstrated that the structural components of the reactor pressure vessel system are within design specifications and that no thermal-hydraulic design or accident analyses criteria are violated. This modification does not result in an unreviewed safety question or Technical Specification change.

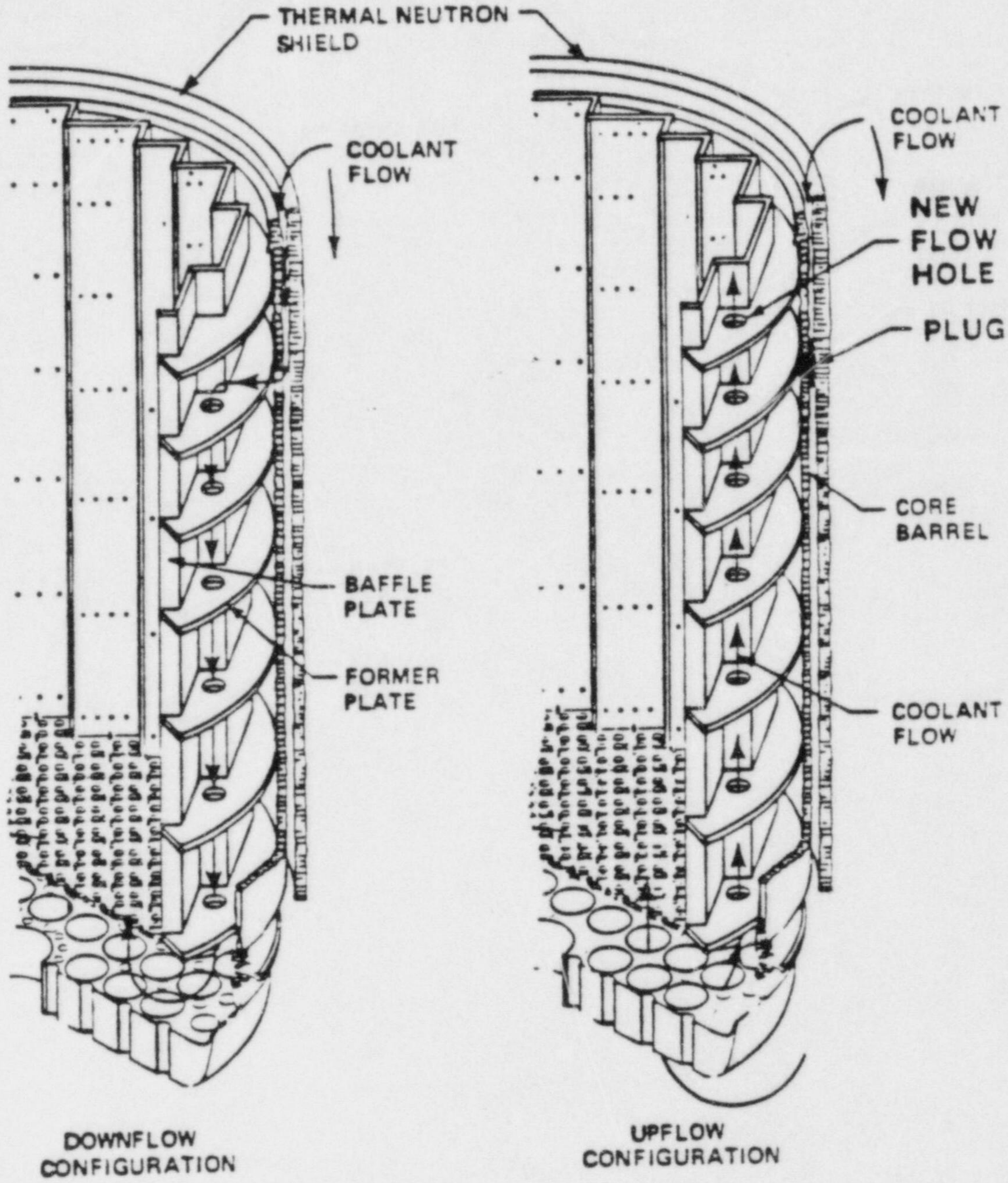
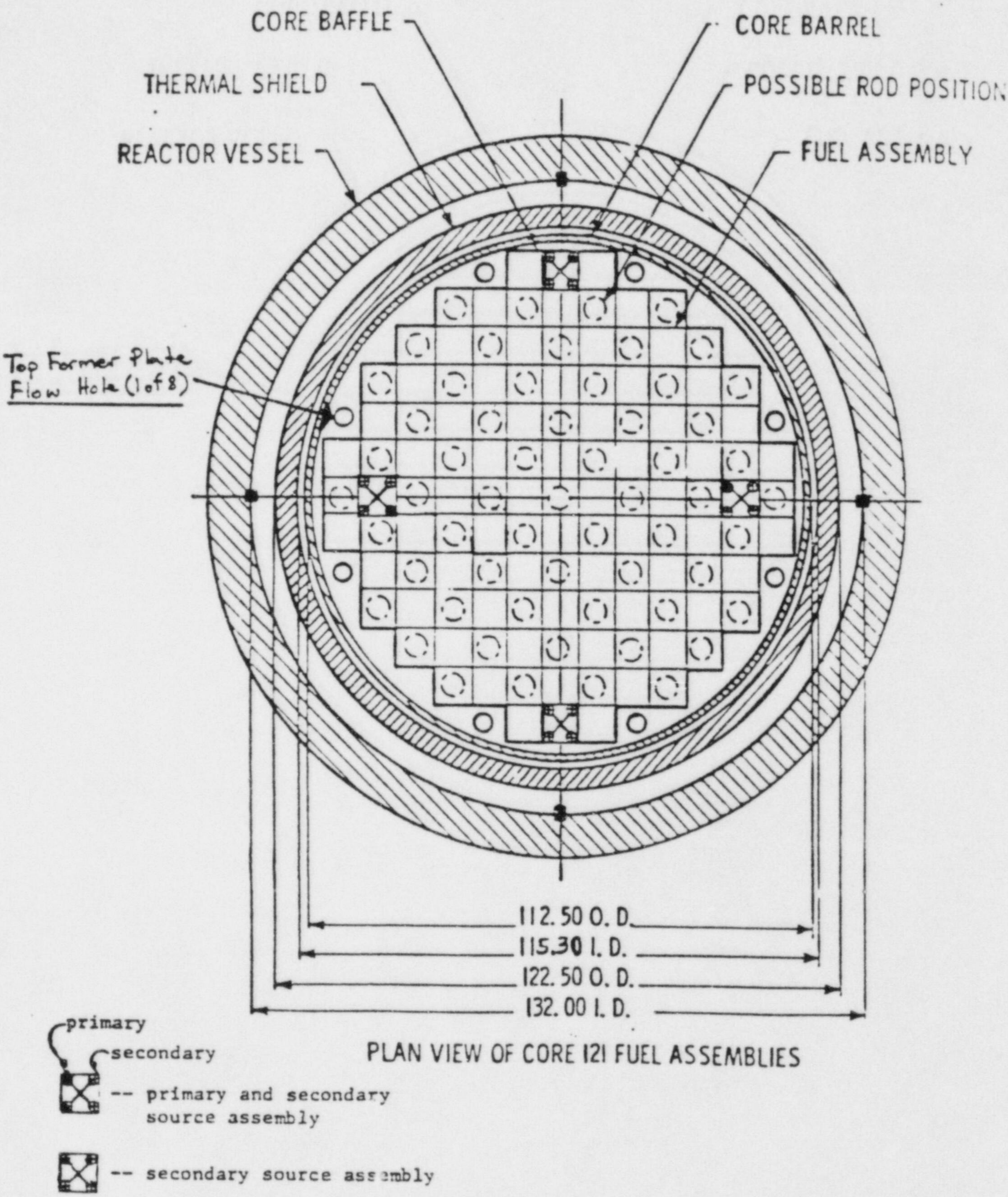


FIGURE 1: FLOW CONVERSION MODIFICATION



REACTOR CORE CROSS SECTION

Figure 2