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AXIAL BLANKET FUEL DESIGN AND DEMONSTRATION

First Semi-Annual
Progress Report
January-September
1980

Prepared for
the United States
Department of Energy

SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT

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**AXIAL BLANKET FUEL
DESIGN AND DEMONSTRATION**

**First Semi-Annual Progress Report:
January-September 1980**

**M. A. Hannah
Project Engineer**

MASTER

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Principal Babcock & Wilcox authors are M. A. Hannah, W. L. Moore, T. N. Ake, A. T. Chiang, G. L. Knock, and J. D. Connell.

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Axial Blanket Fuel Design and Demonstration — First
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Key Words: UO₂ Fuel, Fuel Cycle, Fuel Assembly, Pressurized
Water Reactor, Axial Blanket

ABSTRACT

The Babcock & Wilcox Company, Sacramento Municipal Utility District, and the United States Department of Energy are participating in an axial blanket fuel demonstration program as part of a national effort to improve uranium utilization in light water reactors. The goal of the program is to design and irradiate an entire core of axial blanket fuel assemblies in a commercial pressurized water reactor.

The axial blanket fuel design in this program, which is retrofittable in operating pressurized water reactors, involves replacing the top and bottom of the enriched fuel column with low-enriched (≤ 1.0 wt % ^{235}U) fertile uranium. This repositioning of the fissile inventory in the fuel rod leads to decreased axial leakage and increased discharge burnups in the enriched fuel. The combination of these two phenomena yields both improved uranium utilization and reduced fuel cycle costs without significantly impacting either fuel performance or plant safety.

This report, covering the period from January through September 1980, is the first semi-annual progress report for the program. Various axial blanket fuel designs, with blanket thicknesses from 0 to 10 inches and blanket enrichments from 0.2 to 1.0 wt % ^{235}U , were investigated to determine the relationship between uranium utilization and power peaking. Analyses were performed to assess the nuclear, mechanical, and thermal-hydraulic effects arising from the use of axial blankets.

Four axial blanket lead test assemblies are being fabricated for scheduled irradiation in cycle 5 of Sacramento Municipal Utility District's Rancho Seco pressurized water reactor. Analyses to support licensing cycle 5 are in progress.

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1. INTRODUCTION

Industry attention has been directed toward improving the efficiency of uranium usage and conserving uranium resources¹ since the reprocessing of spent fuel (and hence, the recycling of plutonium and uranium) has been deferred indefinitely. A near-term and readily employable means of improving uranium utilization in pressurized water reactors (PWRs) is to use backfittable axial blankets. This design replaces the top and bottom of the enriched fuel column with low-enriched (≤ 1.0 wt % ^{235}U) pellets while increasing the ^{235}U enrichment of the central pellets in the fuel column. Repositioning of the initial fissile inventory decreases the neutron axial leakage and increases power in the central portion of the fuel rod, which in turn increases the worth of the enriched fuel. The net result is more efficient uranium utilization. Engineering analyses indicate that uranium utilization improvements of up to 4% can be achieved using 9-inch symmetric natural uranium blankets.

The United States Department of Energy (DOE) has initiated a development and demonstration effort involving the Sacramento Municipal Utility District (SMUD) and The Babcock & Wilcox Company (B&W), with the goal of improved uranium utilization through the development and successful operation of axial blanket PWR fuel assemblies in full-core implementation. Studies are being performed under the axial blanket fuel design and demonstration program to evaluate the relationship among axial power peaking, blanket thickness and composition, active fuel length, yellowcake (U_3O_8) savings, and fuel cycle costs in order to develop an axial blanket design. In addition, thermal-hydraulic and mechanical analyses are being conducted to assess the impacts of increased axial power peaking, inherent with backfittable axial blanket designs, on fuel performance and plant safety.

Four axial blanket lead test assemblies (LTAs) are slated for insertion into cycle 5 of SMUD's Rancho Seco reactor. It is further planned that for cycles 6, 7, and 8, all of the fresh fuel introduced into the core will be of the

axial blanket design, so that the entire core will comprise axial blanket fuel assemblies by cycle 8.

This report addresses progress made from January 1 through September 30, 1980; it is the first semi-annual progress report for the DOE/SMUD/B&W program (Contract DE-AC02-79ET34020).

2. SUMMARY

This program will demonstrate improved uranium utilization by developing and successfully operating a PWR-backfittable axial blanket fuel assembly design that meets the operational requirements of current generation standard fuel assemblies. To accomplish this, mechanical and thermal-hydraulic analyses were performed. Efforts during this report period focused on selecting the axial blanket configuration, assessing axial blanket fuel performance characteristics, and evaluating the effects of axial blankets on core thermal-hydraulics, plant safety limits, and operating limits — all for full-core implementation. Work also included the design of four axial blanket lead test assemblies to be irradiated in the Rancho Seco cycle 5 core and the design of the movable incore detector system (MIDS) that monitors the nuclear performance of both an LTA and a standard "non-blanketed" assembly.

For a retrofittable design that retains the same fuel column stack height as a standard assembly, natural uranium blankets (0.711 wt % ^{235}U) in the form of UO_2 pellets exhibited the most favorable balance of improved uranium utilization, reduced separative work requirements, lowest fuel costs, and increased power peaking. A 9-inch natural uranium blanket design provides a 3.9% uranium utilization improvement, a 3.2% reduction in separative work requirements, and a \$1.1 million fuel cost savings per reload batch (\$40/lb U_3O_8 and \$100/kg SWU). Cladding creep collapse, cladding stress and strain, fuel rod internal pressure, fuel rod thermal design, and pellet-cladding interaction (PCI) were evaluated to assess axial blanket fuel performance characteristics. The calculated time to collapse for a 9-inch natural uranium axial blanket fuel rod is less than that for a standard rod (32,000 Vs 35,000 effective full-power hours, EFPH) due to increased axial peaking. However, the projected minimum collapse time is longer than the fuel assembly growth-limited residence time of 30,552 EFPH (43,000 MWd/mtU). Therefore, cladding creep collapse is not a constraint for the axial blanket fuel design.

Since the density and diameter of the fuel pellets and the cladding dimensions are identical for both axial blanket and standard fuel, cladding stress and strain are not significantly affected and the fuel rod thermal design center-line fuel melt limit (CFM) is unchanged. Also, as calculated by the TACO2 computer code², the axial blanket design shows modest (50 psig) increases in fuel rod internal pressure at a fuel assembly average discharge burnup of 35,000 MWd/mtU. The resultant fuel rod internal pressure of 1646 psig has significant margin to the reactor coolant normal operating pressure of 2200 psig. Thus, fission gas release is not a concern in the axial blanket design; neither is PCI a limiting factor since the operational envelope and the availability of volatile fission products are similar to those for a standard design.

The axial blanket fuel design considered in this program is hydraulically and geometrically identical to the standard Mark B fuel. However, the axial peak-increase inherent with the retrofittable axial blanket design does affect core DNBR margin. For example, the beginning-of-life axial peak increases by 12.5% for a 10-inch natural uranium blanket design. This power increase translates to a 22% reduction in DNBR. The 22% penalty could be offset by either reducing the design radial peak by 4.8% to permit a higher axial peak or limiting core power imbalance. Neither method requires changing the thermal-hydraulic design criteria.

A three-dimensional FLAME³ simulation of an axial blanket full-core implementation was performed to assess its impact on plant safety and operating limit envelopes. The resultant envelopes were narrower than those of a standard core but still provided ample latitude for normal power operations.

Four axial blanket LTAs have been designed for insertion in Rancho Seco cycle 5 in 1981. The LTAs are typical Mark B (15 × 15) fuel assemblies with the top and bottom 6 inches of the enriched fuel column replaced by depleted (tails) uranium pellets (0.2 wt % ²³⁵U). The blanket regions contain depleted uranium pellets because they are difficult to model analytically and also provide a measurable contrast to the rest of the fuel assembly for the MIDS to monitor. For ease of identification during manufacture, the axial blanket pellets are 0.20 inch shorter than the enriched Mark B pellets.

The results of the work performed thus far indicate that PWRs can achieve as much as a 3.9% improvement in uranium utilization and an accompanying \$1.1 million reduction in annual fuel costs by using symmetric natural uranium blankets up to 9 inches thick.

3. PROGRAM DESCRIPTION

The objective of the axial blanket program is to demonstrate a significant uranium utilization improvement using axial blankets in an existing PWR. The current program comprises four tasks:

Task 1 — Analytical Studies

Subtask 1: Nuclear Analysis (Blanket Configuration, Safety and Operating Limits)

Subtask 2: Thermal-Hydraulic Analysis (Core Thermal-Hydraulics)

Subtask 3: Mechanical Analysis (Fuel Performance)

Task 2 — Design and Fabrication

Task 3 — LTA Incore Demonstration

Task 4 — Licensing Analysis and Batch Demonstration

The specific objectives, the technical work scope, and the status of each task are described in the following sections.

3.1. Task 1 — Analytical Studies

3.1.1. Objective

Develop an axial blanket fuel configuration for full-core implementation that is consistent with the hardware constraints of the current B&W 15 × 15 fuel assembly design.

3.1.2. Technical Work Scope and Status

3.1.2.1. Subtask 1 — Nuclear Analysis

Evaluate trade-offs in axial power peaking, blanket thickness, blanket material, and active fuel length, U_3O_8 savings and fuel cycle costs for axial blanket thicknesses from 0 to 10 inches and enrichments from 0.2 to 1.0 wt % ^{235}U . Evaluate the effect of full-core axial blanket implementation on plant operating and safety limits. (In progress — see sections 4.1 and 4.5.)

3.1.2.2. Subtask 2 – Thermal-Hydraulic Analysis

Evaluate axial blanket fuel designs with respect to departure from nucleate boiling ratio (DNBR), fuel melt limits, and fuel rod internal pressures. (Complete – see section 4.2.)

3.1.2.3. Subtask 3 – Mechanical Analysis

Evaluate axial blanket fuel designs with respect to PCI sensitivity, cladding creep collapse, and overall fuel rod performance. (Complete – see section 4.3.)

3.2. Task 2 – Design and Fabrication

3.2.1. Objective

Establish manufacturing processes and quality control methods for axial blanket fuel.

3.2.2. Technical Work Scope and Status

Manufacture four axial blanket LTAs. Fabricate axial blanket reload fuel (batches 8, 9, and 10 for cycles 6, 7, and 8, respectively). (In progress – see section 5.)

3.3. Task 3 – LTA Incore Demonstration

3.3.1. Objective

Verify the nuclear methods and models used to design the axial blanket and obtain incore performance data confirming satisfactory operation of the axial blanket fuel assembly.

3.3.2. Technical Work Scope and Status

Design, procure, assemble, and install two movable incore detectors (MIDs) utilizing gadolinium self-powered neutron detectors (SPNDs) in the SMITH Rancho Seco plant prior to cycle 5. Measure axial flux in one blanket LTA and one standard Mark B assembly and benchmark calculations. (In progress – see section 6.)

3.4. Task 4 – Licensing Analysis and Batch Demonstration

3.4.1. Objective

License Rancho Seco cycles 5 through 8 for operation with axial blanket fuel.

3.4.2. Technical Work Scope and Status

1. Produce LTA licensing report to justify cycle 5 operation with the four LTAs inserted. (Planned future work.)
2. Produce the axial blanket fuel topical report to support full-core implementation of axial blanket fuel designs. (Planned future work.)

The topical report justifies implementation of the axial blanket fuel design in the Rancho Seco core and other B&W 177-fuel assembly plants. Mechanical, thermal-hydraulic, and nuclear analyses will be used to compare axial blanket fuel performance to that of standard fuel in order to demonstrate compliance with applicable reactor operating Technical Specifications. (Planned future work.)

4. TASK 1 — ANALYTICAL STUDIES

(M. A. Hannah, T. N. Ake, A. T. Chiang, G. L. Knock)

4.1. Introduction

Nuclear, thermal-hydraulic, and mechanical analyses were performed to develop an axial blanket fuel configuration to be used for full-core implementation. The backfittable design is compatible with existing design constraints of the current B&W Mark B (15×15) fuel assembly.

The nuclear analysis considered several blanket sizes and compositions to determine their relative merits with respect to uranium requirements (utilization), separative work requirements, and fuel costs. Departure from nucleate boiling (DNBR), fuel melt limits, and fuel rod internal pressures were evaluated as parts of the thermal-hydraulic analyses. The mechanical analyses investigated cladding creep collapse, PCI sensitivity, and overall fuel rod performance. The impact on core operation was then assessed for each of the aforementioned factors. The details are presented in the following sections.

4.2. Nuclear Analysis

4.2.1. Blanket Configuration Concept

The backfittable axial blanket design, developed in this study and illustrated in Figure 4-1, does not alter the overall length of the fuel pellet column, but simply redistributes the fissile inventory within that column. Enriched fuel pellets at the top and bottom of the fuel column are replaced with low- ^{235}U pellets, and the enrichment of the central region is increased. The net fissile inventory requirements of the axial blanket configuration are less than those of a standard assembly for the same energy extraction because of decreased axial neutron leakage and better utilization of the enriched fuel.

Beginning-of-life (BOL) axial peaking profiles, presented in Figure 4-1, show significantly lower power in the end regions of the axial blanket design, which in turn translates into reduced axial leakage. The increased axial peaking is

attributable to the blankets at the ends of the fuel column shifting power from these relatively low reactivity regions toward the enriched fuel in the core interior. The shift is less evident at the end of life (EOL) because the plutonium production in the blanket regions and the enriched fuel burnout decrease the reactivity differential between the blanket and the enriched zones.

As shown in Figure 4-2, the axial blanket configuration enables the enriched fuel to be discharged at higher burnups and essentially eliminates the low-burnup (low utilization) enriched end portions of the standard fuel column.

4.2.2. Blanket Configuration Study

The relative merits of different axial blanket lengths and compositions were evaluated using a one-dimensional axial PDQ model.⁴ This model, representing a 142-inch fuel column with a 9-inch axial reflector at each end, was used to simulate fuel from insertion through three cycles of operation. The axial blanket designs were compared to a 3.2 wt % ^{235}U standard assembly with a discharge average exposure of 35,000 Mwd/mtU. The enrichments of the central fuel column of the axial blanket designs were adjusted to produce the same reactivity lifetime (energy extraction) as the standard assembly. Blanket configurations were assessed in terms of (1) axial power peaking, (2) U_3O_8 requirements, (3) separative work requirements (SWU), and (4) fuel costs.

The primary design consideration (as discussed in sections 4.3, 4.4, and 4.5) for axial blanket implementation is increased axial peaking relative to standard fuel assembly designs. Since the greatest difference in peaking occurs at BOL, this is the point at which comparisons were made. Figure 4-3 illustrates the BOL power peaking characteristics for axial blanket materials ranging in thickness from 0 to 10 inches and compositions of 0.2 (depleted), 0.711 (natural), and 1.0 wt % ^{235}U .

The larger the blanket, the greater the peaking increase compared to a standard assembly (0-inch blanket). Likewise, the lower the initial fissile inventory in the blanket, the lower the power in the blanket region and, consequently, the higher the power in the interior. In effect, the axial peaking of axial blanket fuel closely approximates that of a shortened core at BOL.

Plots of percent U_3O_8 savings as a function of symmetric axial blanket size are shown in Figure 4-4 and as a function of BOL axial peaking in Figure 4-5. With respect to axial blanket lengths, the 0.2 wt % ^{235}U material offers the

greatest savings. However, in terms of BOL peaking, depleted uranium becomes advantageous only for configurations with axial peaking increases greater than 9.0%. Natural uranium is the most advantageous material for configurations with axial peaking differences from 5 to 9.0%, below this point, 1.0 wt % ^{235}U is superior.

Separative work savings as a function of symmetric axial blanket sizes and BOL axial peaking are displayed in Figures 4-6 and 4-7, respectively. The 1.0 wt % ^{235}U blanket offers the greatest separative work savings in both blanket size and peaking; natural uranium (0.711 wt % ^{235}U) is second, while depleted (0.2 wt %) blankets offer the least.

Fuel cost savings estimated on \$40/lb U_3O_8 and \$100/kg SWU (separative work unit) are shown in Figure 4-8 plotted against axial blanket lengths and in Figure 4-9 against axial peaking. The savings is based on 60 fuel assemblies, which correspond to one-third core refueling each cycle. Since the cycle length is approximately one year, the fuel cost reduction also corresponds to annual savings.

The natural uranium blankets provide the largest cost savings for any given blanket size and for blanket configurations, with the BOL peaking increase exceeding 7.5%. The 1.0 wt % ^{235}U blankets have the advantage below 7.5% peaking. The depleted uranium blankets (0.2 wt % ^{235}U), as indicated in Figures 4-8 and 4-9, exhibit much smaller fuel cost savings. The large spread between the depleted and other compositions is due primarily to the considerably lower separative work savings component for 0.2 wt % blankets that overshadows the uranium advantage component of the fuel costs.

Fuel costs versus axial peaking are shown in Figure 4-10 for three combinations of \$/lb U_3O_8 and \$/kg SWU. The magnitude of the savings increases with increased fuel costs, but the relationship between 0.711 and 1.0 wt % ^{235}U blankets is not affected significantly. Therefore, the selection of axial blanket composition is relatively insensitive to changes in fuel cost components.

4.2.3. Conclusion

The blanket composition that offers the optimum relationship among savings in uranium requirements, separative work, and fuel cost is natural uranium. Natural uranium provides the greatest fuel cost savings potential (Figure 4-9) and

offers essentially the same U_3O_8 savings as depleted fuel (Figure 4-5) and approaches the separative work savings of 1.0 wt % ^{235}U (Figure 4-7). For configurations with peaking increases below 7.5% neither natural nor 1.0 wt % ^{235}U has an overall advantage. However, since 1.0 wt % ^{235}U is classified as "special nuclear material" and natural uranium is not, differences in procurement, processing, storage, and fabrication costs (not included in Figures 4-8, 4-9, and 4-10) could reduce the fuel cost advantages of 1.0 wt % ^{235}U . Therefore natural uranium blanket configurations are recommended for fuel implementation.

4.3. Thermal-Hydraulic Analysis

4.3.1. Core Thermal-Hydraulics

The backfittable axial blanket fuel design considered in this program and the standard Mark B fuel design employ the same assembly hardware; therefore, the two are hydraulically and geometrically identical. The fuel pellet column length is the same in both designs and, consequently, the hydraulic loads for both are also identical.

The core thermal-hydraulic analysis is based on an assumed reference power distribution. This assumption is ensured by applying appropriate criteria to a fuel cycle design and to both plant operating and reactor protection limits. These limits constrain the allowable power distributions to maintain the initial DNBR assumed for accident analyses and to guarantee that the 1.3 design DNBR is not violated during steady-state operation. Since the power peaking characteristics of axial blanket fuel differ significantly (section 4.2.2) from those of standard fuel, the effects of these differences on DNBR margin were evaluated.

4.3.2. DNBR Evaluation

DNBR evaluations for the B&W Mark B (15 × 15) 177-fuel assembly core were performed for the 112% design overpower conditions (shown in Table 4-1) using the CHATA⁵ and TEMP⁶ computer codes. By using the reference design peaking distribution* for standard Mark B fuel at 112% design overpower, a minimum DNBR of

*1.714 FA_h (radial × local) × 1.5 F_z (axial).

1.74 is obtained. When the increased axial peak inherent with the retrofittable axial blanket design is combined with the 1.714 FA_h , the DNBR is reduced.

Table 4-2 lists the minimum DNBR values that result from the axial peaks corresponding to 6- and 10-inch symmetric natural uranium blankets. The 12% axial peaking increase for 10-inch blankets gives a 22% DNBR reduction; for 6-inch blankets, 10%. Table 4-2 also shows that by reducing the design FA_h peaking factor, the core thermal margin that existed for the base design (DNBR = 1.74) can be re-established. DNBR is more sensitive to hot channel power (FA_h) than to axial (F_z) power distributions. Therefore, a 4.87% FA_h reduction offsets the 12% axial peaking increase for 10-inch blankets, and only a 2.30% reduction offsets the effect of a 6-inch blanket. The magnitude of these changes indicates that in determining the safety and operating limits, control to the same peaking criteria is conservative for the axial blanket design. This conservative assumption does not restrict the safety limits (section 4.5). However, should this conservative assumption be too restrictive, the total peak can be increased by reducing the radial peak (see Table 4-2). Since current Mark B reload core designs typically maintain more than a 10% peaking margin to the design FA_h of 1.714, even the 4.87% decrease for the 10-inch blanket design can be easily accommodated.

4.4. Mechanical Analysis

4.4.1. Fuel Performance

The objective of these analyses was to determine the effects of axial blanket fuel rod designs on Mark B fuel performance, i.e., cladding creep collapse, fission gas release, fuel rod thermal design, and PCI.

4.4.2. Creep Collapse Analysis

A creep collapse analysis was performed for both standard and 9-inch (natural uranium) axial blanket fuel assemblies. Ovality changes in fuel rod cladding were calculated with the CROV computer code⁷, which predicts the collapse behavior of a fuel rod. The TACO2 code² provided fuel rod internal pressures and cladding temperatures for input to CROV. The following conservative assumptions were used in modeling the fuel rod:

- Zero fission gas release.
- Minimum initial fill gas pressure.
- The lower tolerance limit (LTL) for cladding thickness.
- The upper tolerance limit (UTL) for critical cladding ovality.
- Worst-case power history for creep collapse.

The results of the CROV analysis (shown in Table 4-3) indicate that the time to collapse for a 9-inch axial blanket fuel rod is >32,000 EFPH and for standard assemblies, >35,000 EFPH. Fuel assembly growth limits the Mark B assembly to a burnup of 43,000 MWd/mtU, which corresponds to an equivalent residence time of 29,280 EFPH.¹ Even though the time to collapse is shortened in the axial blanket design, the conservative projected minimum collapse time is still longer than the expected maximum residence time. Therefore, cladding creep collapse is not a constraint for the axial blanket fuel rod design.

4.4.3. Cladding Stress and Strain

The pellet and cladding dimensions given in Table 4-3 are the same for both the axial blanket and standard fuel assembly designs. Stress is not a burnup-limiting phenomenon. Based on cladding strain, the Mark B fuel assembly has a burnup limit of 56,000 MWd/mtU¹, which is greater than the expected maximum burnup of 43,000 MWd/mtU. Therefore, cladding stress and strain do not significantly impact the axial blanket design.

4.4.4. Fuel Rod Internal Pressure

A fuel rod internal pressure analysis was also performed for axial blanket fuel using the TACO2 computer code.² The TACO2 results are the most realistic predictions available for fission gas release and fuel rod internal pressure calculations during fuel residence. A linear heat rate envelope for the Mark B 2772-MWt core, coupled with axial power and burnup shapes for the axial blanket design, were used in the TACO2 analysis. The results shown in Table 4-4 indicate that the 10-inch axial blanket design (the worst case considered) results in a slight increase (50 psig) in internal pin pressure at a discharge burnup of 35,000 MWd/mtU. Table 4-4 also shows that adequate margin from the system pressure also exists for burnups up to 50,000 MWd/mtU, which is greater than the limiting burnup of 43,000 MWd/mtU. Therefore, significant margin still remains before exceeding the reactor coolant operating pressure of 2200 psig, indicating that fission gas release is not a limiting concern for the axial blanket design.

4.4.5. Fuel Rod Thermal Design

The density and diameter of the fuel pellets in the axial blanket fuel design are identical to those of the standard Mark B design. Since the fuel density and the gap between the pellets and cladding are the primary factors affecting the heat transfer from the pellets, the average temperatures and the centerline fuel melt limit (CFM, typically 20.4 kW/ft for the Mark B core) are unchanged. Therefore, there are no significant differences in fuel thermal performance limits between the axial blanket and standard fuel contained in Rancho Seco.

The maximum LHR capability and the average fuel temperature for cycle 5 blanket-ed and non-blanketed fuel are shown in Table 4-5. LHR capabilities are based on centerline fuel melt and were established using the TAFY3 code⁸ with fuel densification to 96.5% of theoretical density (TD).

4.4.6. Pellet-Cladding Interaction (PCI)

PCI is generally believed to be dependent on the availability of volatile fission products (iodine and cesium) and local changes in cladding stress and strain caused by certain power maneuvers. As discussed in section 4.4.4, the fission gas release during steady-state operation increases only slightly for the axial blanket design. Simulated load-follow maneuvers using the 9-inch axial blanket design show less than one kW/ft difference from the standard design. Therefore, axial blanket implementation will not affect PCI performance.

4.5. Safety and Operating Limits

Allowable operating conditions for B&W cores are defined by Technical Specifications, which include safety limits (reactor protection system, RPS) and operating limits.

4.5.1. Safety Limits

The plant is operated within RPS imbalance limits (see Figure 4-11) to prevent DNB or centerline fuel melt (CFM) during steady-state and transient conditions. The RPS automatically shuts down the reactor if the specified limits are reached. To establish the RPS limits, power peaking is correlated with core power imbalance (defined below) for various CRA positions, xenon conditions, and cycle lifetimes:

$$\% \text{ imbalance} = \frac{\left(\begin{array}{c} \text{power} \\ \text{in} \\ \text{top} \end{array} \right) - \left(\begin{array}{c} \text{power} \\ \text{in} \\ \text{bottom} \end{array} \right)}{\text{rated power}} \times 100$$

In addition to the RPS imbalance limits, a 1% $\Delta k/k$ shutdown margin is ensured through control rod assembly (CRA) position limits.

4.5.2. Operating Limits

Operating limits define the permissible ranges for CRA positions, axial power shaping rod (APSR) positions, and power imbalance during normal operation and are based on analyses of emergency core cooling system (ECCS) performance during a loss-of-coolant accident (LOCA). These operating limits ensure plant operation within specified LHRs that guarantee compliance with the criteria of 10 CFR 50, Appendix K.

Unlike the RPS limits, a violation of the operating limits does not automatically shut down the reactor; however, it must be returned to its permissible operating envelope within a specified time frame — normally two hours. Examples of these limits are provided in Figures 4-12, 4-13, and 4-14.

The DNB, CFM, and LOCA kW/ft criteria are compared in Figure 4-15. For negative imbalance configurations, the safety limits are normally set by the CFM criterion. Since the DNB is more restrictive at the top of the core, configurations resulting in a positive imbalance are usually established by the DNB criterion. The LOCA limit is more restrictive than either the CFM or DNB limits, which are discussed in detail in sections 4.4.5 and 4.3.2, respectively.

In this analysis, a three-dimensional core simulation was performed to assess the affect of axial blanket fuel implementation on plant safety and operating envelopes using the same design criteria as those used for a typical reload core of standard Mark B fuel.

4.5.3. Three-Dimensional Core Simulation

The nodal code, FLAME, was used to model the core in quarter-core geometry with one radial node and 32 axial nodes per assembly.³ Axial blanket cycles were evaluated using the same fuel loading patterns, CRA positions, and lumped burnable poison (LBP) boron-10 concentrations as those actually employed in Rancho Seco cycles 3, 4, and 5. Cycle 4 of Rancho Seco is the current operating cycle, and cycle 5 is the next reload (described in section 6). The characteristics

of both the axial blanket implementation (ABI) and standard fuel cycle 5 (chosen as the reference core) are given in Table 4-6 and Figure 4-16. The ABI fuel enrichment was increased to obtain reactivity lifetimes (burnups) comparable to those of the reference core with the exception of 13 batch 1 assemblies not modeled as blanketed fuel.

The axial blanket fuel design calculations for the ABI cycle 5 contained symmetric 9-inch axial blankets of natural uranium, with the total pellet stack height (enriched plus blankets) unchanged from that of the standard Mark B fuel. The 9-inch blanket was studied for two reasons: (1) the FLAME nodal spacing in the base cycle was 4.5 inches, and (2) 9-inch blankets are a practical upper bound in terms of axial peaking. By maintaining the same geometry and fuel cycle designs, three-dimensional core analyses of the two cycle 5 cores (ABI and reference) provide direct comparisons of safety and operating limits from which the effect(s) of the axial blanket implementation can be accurately and realistically determined.

4.5.4. Discussion

With the exception of the shutdown margin criterion, the safety and operating limits are based on power peaking considerations. As discussed in section 4.1.2, the overall effect of blanketed fuel on power peaking is similar to what would result from shortening the core. The radial peaks are virtually unchanged by the blanket (Figure 4-17). The only appreciable differences are attributable to the presence of partially inserted CRAs (group 7) and the APSRs, both of which trigger different axial responses in the two cores.

The group 7 CRAs are 12.5% inserted from the top into the assemblies at core locations H10, H14, and N12. Since they are inserted in the low-power blanket region, they have less impact on the power in these assemblies for the ABI core. Conversely, the assembly power in core location L12 is lower because the APSR has greater worth in the central region of the ABI core.

Consistent with the PDQ results reported in section 4.1.2, the three-dimensional FLAME analysis showed greater peaking in the axial blanket core. Table 4-7 shows that the axial peaking difference decreased from 12.5% at the beginning of the cycle to 6.8% at the end of the cycle.

Core-averaged axial power shapes are shown in Figures 4-18 and 4-19 at 4 and 304 EFPD, respectively. In both instances, the ABI has a higher axial peak relative to the reference core, but the difference between the two cores decreases with cycle exposure. In both instances, the ABI core has a higher axial peak relative to the reference core, but the difference between the two cores decreases with cycle exposure.

As shown in Table 4-8, the RPS safety limits were set at the beginning of the cycle, which was the most constraining time in cycle life. The use of 9-inch axial blankets results in a decrease of 9% imbalance for the RPS negative imbalance limits. This decrease is seen because the increased axial peaking in axial blanket cores causes the CFM limit to be reached at less negative imbalance conditions. It is very difficult to induce a high positive imbalance in a PWR reload core by using CRAs and xenon; therefore, neither the CFM nor the DNB limits were encountered for positive imbalance conditions in either core. The positive imbalance limits for both cores were set by conservative extrapolations of available positive imbalance cases. The RPS envelopes for the ABI and reference cycle 5 cores are both given in Figure 4-11. The ABI operating envelope is narrower than that of the reference core but still provides ample latitude for normal power operations.

The operating limits for the ABI and reference cycle 5 cores are shown in Table 4-9 together with the previous actual operating limits for cycles 2, 3, and 4. As with the safety limits, the higher axial peaking in the blanketed core results in more restrictive limits. However, since the limits of the ABI cycle 5 core are comparable to those of previous cycles, which operated satisfactorily, they will allow acceptable full-power operation. Wider operational limits can be obtained, if necessary, by using smaller blankets (lower peaking).

The total and differential CRA worths for the ABI and reference cores were also compared — the total worth of the CRAs for both cores is the same, which is to be expected since the radial power distributions are similar. A graph of inserted reactivity as a function of insertion is shown in Figure 4-20. The CRAs have essentially no worth in the low-power blanketed regions of the core; their differential worth in the center portions is higher, resulting in no net change in total worth for the ABI core.

The CRA insertion limits imposed to maintain a 1% $\Delta k/k$ shutdown margin with the highest worth CRA stuck in a fully withdrawn position are essentially the same for both cores. However, deeper insertion of CRAs may be necessary to maintain a quick reactivity change capability ("bite"). For instance, to maintain a CRA positive bite in the reference core, 7% insertion of group 7 is required, while 15% insertion is necessary in the ABI core.

In addition, the relative xenon stabilities for both the ABI and reference cores were evaluated by performing a 100-60-100% full power transient simulation. The resultant imbalance oscillation after full-power recovery was followed for 52 hours and is plotted in Figure 4-21. The period of the imbalance of oscillation is the same for both cores, but the amplitude in the ABI core is reduced. This increased stability to axial xenon oscillations is consistent with results for shorter cores.

Table 4-1. Thermal-Hydraulic Design Conditions

Design power level, MWt	2772
System pressure, psia	2200
Reactor coolant flow, % design flow	104.9
Vessel inlet/outlet coolant temp, 100% power, F	557.7/606.3
Reference $F\Delta_h$ (radial $\times$ local) peaking factor	1.71
Reference F_z (axial) power shape	1.5 cosine with tails
Hot channel factors	
Enthalpy rise (F_q)	1.011
Heat flux (F_q)	1.014
Flow area	0.98
Active fuel length, in.	141.8
Avg heat flux, 100% power, Btu/h-ft ²	1.9E05 ^(a)
Max heat flux, 100% power, Btu/h-ft ²	494E05 ^(a)
CHF correlation	BAW-2
Minimum DNBR (% power)	1.74 (112%)

(a) With thermally expanded fuel rod OD of 0.43075 inch.

Table 4-2. Minimum DNBR Vs Total Peaking

Blanket size, in.	$F\Delta_h$	F_z	Total peak ($F\Delta_h \times F_z$)	Minimum DNBR
0 (ref)	1.714	1.50	2.57	1.74
6	1.714	1.59	2.73	1.62
10	1.714	1.68	3.39	1.52
6	1.675	1.59	2.66	1.74
10	1.630	1.68	2.74	1.74

Table 4-3. Fuel Rod Parameters and Dimensions

<u>Fuel assembly type</u>	<u>Mark B</u>	<u>Mark B axial blanket</u>
Fuel rod OD, in.	0.430	0.430
Fuel rod ID, in.	0.377	0.377
Flexible spacers, type	Spring	Spring
Rigid spacers, type	Zircaloy-4	Zircaloy-4
Undensified fuel length, in.	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3686	0.3686
Fuel pellet initial density (mean spec), % TD	95.0	95.0
Estimated residence time, EFPH	29,280	29,280
Cladding collapse time, EFPH	>35,000	>32,000

Table 4-4. Fuel Rod Internal Pressure Results

<u>Operation</u>	<u>Internal rod pressure, psig</u>		
	<u>Standard assembly</u>	<u>6-in. blanket</u>	<u>10-in. blanket</u>
After three cycles, 35,000 MWd/mtU	1598	1600	1646
Projected value at 50,000 MWd/mtU	1949	1969	1994

Table 4-5. Fuel Thermal Analysis Parameters

	<u>Standard Mark B</u>	<u>10 in. blanket</u>
Initial density, % TD	95	95
Pellet diameter, in.	0.3686	0.3686
Nominal stack height, in.	141.8	141.8
Nominal LHR, kW/ft at 2772 MWt	6.27	6.27
Avg fuel temperature at nominal LHR, F	1442	1442
LHR capability, kW/ft to centerline fuel melt	20.4	20.4
Densification data, % at 4000 MWd/mtU	96.5	96.5

Table 4-6. Fuel Cycle Characteristics,
Rancho Seco Cycle 5

2772 MWt
325 EFPD design length
Feed-and-bleed operation
APSR withdrawal at 300 EFPD

Table 4-7. Core Average Axial Peaking for
Blanketed and Non-Blanketed Core

<u>EFPD</u>	<u>References</u>	<u>ABI</u>	<u>% diff</u>
4	1.23	1.38	12.2
100	1.16	1.27	9.5
200	1.14	1.23	7.9
300	1.12	1.22	8.9
325	1.18	1.26	6.8

Table 4-8. RPS Safety Limits - 4 EFPD, 9-Inch
Natural U Blanket, 112% FP

	<u>Reference</u>	<u>ABI</u>	<u>Net change</u>
+ % imbalance	35	32	3
- % imbalance	-40	-31	-9

Table 4-9. BOL Operating Limits (LOCA) 102% FP

		<u>Cycle 5</u>	<u>Rancho Seco</u>		
	<u>Cy5 ABI</u>	<u>reference</u>	<u>Cycle 4</u>	<u>Cycle 3</u>	<u>Cycle 2</u>
+ imbalance	18	24	24	23	12.2
- imbalance	-11	-19	-18	-14	-12.2
Rod index, % wd	280-300	280-300	280-300	380-300	283.8-300
APSRs, % wd	12-32	8-37	6-27	8-32	0-22.3

Figure 4-1. Axial Power Peaking Profiles

Figure 4-2. Discharge Exposure Profiles

	<u>Standard</u>	<u>Axial Blanket</u>
$\text{U}_3\text{O}_8/\text{FA}$, lb	7184	6965
U/FA, kg	463.6	463.6
BURNUP, MWd/mtU	35,000	35,000
UTILIZATION,	12.38	12.78
MWY/STU $_{308}$		

Figure 4-3. Beginning-of-Life Axial Power Peaking Vs Symmetric Axial Blanket Lengths

Figure 4-4. Uranium Requirements Vs Symmetric Axial Blanket Lengths

Figure 4-5. Uranium Requirements Vs BOL Axial Power Peaking

Figure 4-6. Separative Work Requirements Vs Axial Blanket Lengths

Figure 4-7. Separative Work Requirements Vs BOL Axial Power Peaking

Figure 4-8. Fuel Costs Vs Symmetric Axial Blanket Lengths

Figure 4-9. Fuel Cost Vs BOL Axial Power Peaking

Figure 4-10. Fuel Costs Vs BOL Axial Power Peaking

Figure 4-11. Reactor Protection System Safety Limits, Core Power Imbalance

Figure 4-12. Core Power Imbalance Operating Limits

Figure 4-13. Rod Index Vs Power Level

Figure 4-14. APSR Withdrawal Operating Limits

Figure 4-15. LOCA, DNB, CFM Limit Curves

Figure 4-16. Rancho Seco Cycle 5 Core Loading Map

	8	9	10	11	12	13	14	15
H	1	7	5	7	5	5	6	6
K	7	5	5	5	7	5	7	6
L	5	5	5	7	1	7	6	6
M	7	5	7	1	7	5	6	
N	5	7	1	7	5	7	6	
O	5	5	7	5	7	6		
P	6	7	6	6	6			
R	6	6	6					

Batch	No.	Enrichment, wt % ²³⁵ U		Remarks
		Base	ABI	
1	13	2.01	2.01	Once burned
5	56	3.04	3.29	Twice burned
6	52	3.21	3.48	Once burned
7	56	3.14	3.40	Fresh

Figure 4-17. Beginning of Cycle Radial Power Peaking Comparison

	8	9	10	11	12	13	14	15
H	0.95	1.22	1.04	1.28	1.04	1.09	1.06	0.62
	0.95	1.24	1.05 CRA	1.29	1.04	1.08	1.08 CRA	0.63
K		1.03	1.16	1.17	1.25	1.16	1.16	0.60
		1.04	1.16	1.16	1.24	1.15	1.19	0.61
L			1.19	1.25	0.91	1.24	0.99	0.46
			1.18	1.24	0.88 APSR	1.23	1.00	0.46
M				0.99	1.24	1.08	0.72	
				0.97	1.24	1.09	0.72	
N					0.98	1.04	0.49	
					0.99 CRA	1.05	0.50	
O						0.58 0.59	Standard Axial Blanket } Assembly RPD	
P								
R								

Figure 4-18. Core Average Axial Power Shapes, 4 EFPD

Figure 4-19. Core Average Axial Power Shapes, 304 EFPD

Figure 4-20. Inserted Worth Vs Rod Index

Figure 4-21. 304 EFPD Xenon Transient, 100-60-100% FP With Recovery at Time of Peak Xenon

5. TASK 2 — DESIGN AND FABRICATION

(W. L. Moore)

5.1. Introduction

The four axial blanket LTAs slated for insertion in SMUD's Rancho Seco cycle 5 core consist of a standard B&W Mark B fuel assembly in which the top and bottom 6 inches of enriched fuel are replaced with depleted fuel (tails) in all 208 rods. Figure 5-1 gives the significant overall dimensions of the fuel assembly and indicates the location of the axial blankets. All hardware used in the fuel assembly is common to both the LTAs and the standard fuel assemblies that make up the balance of the reload.

5.2. Discussion

Figure 5-2 is a diagram of the axial blanket fuel rod. The standard rod contains a nominal 141.8 inches of 3.14 wt % ^{235}U enriched fuel; the LTA fuel rods contain 129.8 inches of the same fuel situated between two 6-inch sections of depleted uranium fuel (0.20 wt % ^{235}U). The nominal density and diameter of both the depleted and enriched UO_2 pellets are 95.0% TD and 0.3686 inch, respectively; the enriched and depleted pellets are 0.6 and 0.4 inch long. The difference between the two pellet lengths is for ease of identification during manufacture.

Characterization measurements will be taken on a total of 40 axial blanket fuel rods and will consist of the following:

1. Fuel rod tube ultrasonic trace data for ID, OD, wall thickness, and eccentricity. (ID, OD, and wall thickness are direct measurements, while eccentricity is obtained by subtracting ID from OD.)
2. As-built axial blanket and enriched fuel stack heights.
3. Pellet density and diameter for three pellets on each side of the juncture of the enriched and axial blanket stacks.

Thirty-two characterized rods are to be located on the outside of each LTA - at each corner and the mid-face position, as shown in Figure 5-3. One LTA will be monitored by a movable incore detector (section 6); this one will contain eight characterized rods around the instrument tube (Figure 5-3a). In addition, these eight rods are to be x-rayed over their entire length to correlate axial traces from the movable incore detector with the as-built positions of the axial blankets.

Figure 5-1. Axial Blanket LTA Fuel Assembly

Note: Dimensions in inches.

Figure 5-2. Axial Blanket LTA Fuel Rod

NOTE: DIMENSIONS IN INCHES.

Figure 5-3. LTA Characterized Fuel Rod Locations

a. LTA CONTAINING MIDS

b. THREE LTA'S

GUIDE TUBE

INSTRUMENT TUBE

FUEL ROD

FUEL ROD (CHARACTERIZED)

FUEL ROD (CHARACTERIZED AND ADDITIONAL
CONTROL PLACED ON FABRICATION OF
AXIAL BLANKET)

6. TASK 3 — LTA INCORE DEMONSTRATION

(M. A. Hannah)

6.1. Introduction

Four LTAs will be irradiated in cycles 5, 6, and 7 of Rancho Seco; their planned locations in cycle 5 are shown in Figure 6-1. The axial flux shape in one LTA and a symmetric standard assembly will be monitored with two movable incore gadolinium self-powered neutron detectors (SPNDs) in conjunction with fixed incore rhodium SPNDs. The movable detectors are necessary to provide more accurate axial blanket performance since the fixed incore detectors are designed to measure the neutron flux in the core interior and do not extend into the blanketed regions (Figure 6-2). Plans include using the movable SPNDs after every 30 EFPD of operation to obtain detailed neutron flux data coincident with discrete axial and radial measurements provided by the fixed incore SPNDs. The flux information will be used for comparison to the analytical predictions used in designing and licensing axial blanket fuel.

6.2. Movable Incore Detector System

The movable incore detector system (MIDS) comprises three major components: movable incore detectors, detector drive units, and drive control and signal readout equipment. One detector and one drive are designated for each fuel assembly intended for neutron flux monitoring. The drive units are located in the incore instrument area inside the reactor containment; both drives (one for each detector) are actuated from a control and readout station outside the containment. On command from the control station, a drive unit inserts its movable incore detector into the active core region through the calibration tube of the fixed incore detector (Figure 6-3). Guided by the calibration tube, the MID passes upward through the core, automatically reverses its travel at the top, and then withdraws from the core to an inactive position approximately 10 feet below the bottom of the core. During both the insertion and removal traverses, signals from the movable detectors are recorded on a strip chart at the readout station.

6.3. Movable Incore Detector

The MID (shown in Figure 6-4) is a self-powered neutron flux sensor connected only to the signal conditioning equipment. The neutron-induced signal is produced entirely within the 1-inch gadolinium emitter, which measures the absorption of thermal neutrons. The 2200 m/second neutron capture cross section of gadolinium is 48,000 barns. The dominant neutron reaction is as follows:

The neutron induced signal is generated in the following sequence:

1. A thermal neutron is captured within the gadolinium emitter.
2. The gadolinium nucleus that absorbed the thermal neutron releases captures gammas. An average of 337 gammas are released for every 100 neutron captures, with energies ranging from 100 keV to 7.5 MeV.
3. A small fraction (1 to 2%) of the capture gammas undergo either Compton scattering or photoelectric interactions to produce energetic electrons. The Compton effect accounts for ~80% of the total interactions.
4. Many of the energetic electrons produced in step 3 escape from the gadolinium emitter, cross the aluminum oxide insulation (Figure 6-4), and are collected at the detector sheath.

The electron flow in step 4 is the positive current measured with external circuitry and is directly proportional to the incident neutron flux. In addition, a background signal due to core gammas is generated in the detector leadwire and is directly proportional to the length of the leadwire within the core. The signal from the background wire, also produced by core gammas, is used to cancel out the background wire's contribution to the detector signal. Since the neutron signal is generated within a 1-inch span of the core, the moving detector's signal is a true spatial map of the thermal neutron flux within the core. This neutron capture and gamma production is an instantaneous event not affected by the velocity of the detector through the core. A typical axial trace using the MIDS is shown in Figure 6-5.

6.4. Design of Rancho Seco Cycle 5

The symmetric core locations for the four axial blanket LTAs to be irradiated in cycle 5 of Rancho Seco are B7, G14, P9, and K2 (Figure 6-1). The lead test assembly in location B7 and the standard assembly in location G2 contain fixed rhodium SPND string numbers 31 and 36, respectively, in addition to movable gadolinium SPNDs.

Cycle 5 was designed for a 325-EFPD (13 month) cycle length using an in-out-in LBP shuffle scheme. This configuration exhibits improved neutron economy over standard out-in-in schemes because it reduces core radial leakage by placing the fresh fuel in the interior of the core, permitting enrichment savings.

The fuel shuffle scheme and batch designations for cycle 5 are shown in Figure 6-6. The feedbatch of 56 assemblies (including the four lead test assemblies) of 3.14 wt % ^{235}U are loaded in a "broken checkboard" pattern in the core interior. The once-burned (batch 6) fuel is placed on the core periphery. Thirteen batch 1 fuel assemblies, discharged at the end of cycle 1, together with twice-burned (batch 5) fuel are positioned in the interior region of the core. LBP clusters are loaded in fresh fuel assemblies to control power peaking and to reduce soluble boron concentration requirements at the beginning of the cycle. The LBP locations and concentrations are given in Figure 6-7.

Calculated assembly relative power at the beginning and end of cycle 5 are shown in Figure 6-8. The power distributions were generated by the PDQ computer code⁴ with two-dimensional (X-Y), quarter-core geometry and a pin-by-pin representation of the core with thermal-hydraulic feedback (B&W's standard design model). The LTAs are symmetric to location K14 in Figure 6-8 and maintain at least 31% margin to the design radial peaking limit during cycle 5. Therefore, the LTAs are not limiting assemblies during cycle 5.

The assembly burnups given by PDQ calculations for Rancho Seco cycle 5 are presented in Figure 6-9 for the beginning and end of the cycle. The LTAs will accumulate an exposure of 13,037 MWd/mtU for an average assembly relative power of 1.15.

6.5. Conclusions

Cycle 5 (325 EFPD, 13 months) was designed to meet normal operating design criteria. The inclusion of four lead test assemblies has a negligible effect on core operation.

Figure 6-1. Lead Test Assembly Locations

Figure 6-2. Fixed SPND Positioning Relative to LTA Blanket Regions

FIXED RHODIUM INCORE. AXIAL BLANKET LEAD TEST ASSEMBLY
SPND STRING

Figure 6-3. Instrument Tube Cross Section

Figure 6-4. Twin-Lead Gadolinium Detector

Figure 6-5. Typical MIDS Axial Trace

- Rn SPND LOCATIONS
- Rn SPND POWER PROFILE
- - MIDS Gd TRACE

Figure 6-6. Full Core Shuffle, Rancho Seco Cycle 5

A						M4 6	K2 6	G8 6	K14 6	M12 6					
B				L5 6	L3 6	N3 6	LTA 7	E8 6	7	N13 6	L13 6	L11 6			
C			H3 6	7	B4 5	7	A7 5	R8 5	A9 5	7	B12 5	7	C8 6		
D		E10 6	7	B6 5	7	(Cyl) C5 1C	7	B8 5	7	(Cyl) C11 1C	7	F14 5	7	E6 6	
E		C10 6	D2 5	7	(Cyl) E5 1C	7	B5 5	7	B11 5	7	(Cyl) E11 1C	7	D14 5	C6 6	
F	D11 6	C12 6	7	(Cyl) E3 1C	7	O13 5	A6 5	B10 5	A10 5	O3 5	7	(Cyl) E13 1C	7	C4 6	D5 6
G	B9 6	7	G1 5	7	E2 5	F1 5	L10 5	7	L6 5	F15 5	E14 5	7	G15 5	LTA 7	B7 6
H	H7 6	H5 6	H15 5	H2 5	7	F2 5	7	(Cyl) G9 1C	7	L14 5	7	H14 5	H1 5	H11 6	H9 6
K	P9 6	LTA 7	K1 5	7	M2 5	L1 5	F10 5	7	F6 5	L15 5	M14 5	7	K15 5	7	P7 6
L	N11 6	O12 6	7	(Cyl) M3 1C	7	C13 5	R6 5	P6 5	R10 5	C3 5	7	(Cyl) M13 1C	7	O4 6	N5 6
M		O10 6	N2 5	7	(Cyl) M5 1C	7	P5 5	7	P11 5	7	(Cyl) M11 1C	7	N14 5	O6 6	
N		M10 6	7	L2 5	7	(Cyl) O5 1C	7	P8 5	7	(Cyl) O11 1C	7	P10 5	7	M6 6	
O			O8 6	7	P4 5	7	R7 5	A8 5	R9 5	7	P12 5	7	H13 6		
P				F5 6	F3 6	D3 6	7	M8 6	LTA 7	D13 6	F13 6	F11 6			
R						E4 6	G2 6	K8 6	G14 6	E12 6					
		2	3	4	5	6	7	8	9	10	11	12	13	14	15

XX
Y

Previous Cycle Location
Batch Number

LTA = Axial Blanket Lead
Test Assemblies

Figure 6-7. LBP Positions, Rancho Seco Cycle 5

	8	9	10	11	12	13	14	15
H		0.80		1.10				
K	0.80				1.10		0.50	
L				1.10		0.80		
M	1.10		1.10		0.50			
N		1.10		0.50		0.20		
O			0.80		0.20			
P		LTA 0.50						
R								

<u>No. of BFRAs</u>	<u>Concentration wt % B₄C in Al₂O₃</u>
-------------------------	---

8	0.20
16	0.50
12	0.80
20	1.10

Figure 6-8. Two-Dimensional Relative Power Distribution,
Rancho Seco Cycle 5

	8	9	10	11	12	13	14	15
H	0.92 0.91	1.24 1.20	1.07 1.00	1.27 1.29	1.03 1.03	1.11 1.04	1.14 1.01	0.66 0.65
K		1.03 0.96	1.18 1.09	1.13 1.11	1.23 1.33	1.15 1.09	1.17 1.11	0.64 0.64
L			1.18 1.12	1.23 1.34	0.85 1.05	1.23 1.27	1.00 0.94	0.48 0.50
M				0.93 1.01	1.26 1.30	1.08 1.04	0.74 0.74	
N					1.02 0.98	1.02 0.99	0.50 0.52	
O						0.59 0.61	BOC5, RPD - APSRs Inserted EOC5, RPD - APSRs Withdrawn	
P								
R

XXX Relative Power Density

Symmetric location of LTAs

Figure 6-9. Assembly Burnup Distribution, Cycle 5

	8	9	10	11	12	13	14	15
H	18,058 28,610	0 14,274	22,147 34,024	0 14,749	25,406 37,104	16,871 29,034	13,681 25,820	14,099 21,462
K		24,929 36,296	13,180 26,059	18,226 30,953	0 14,470	16,285 28,929	0 13,037	13,202 20,422
L			15,190 28,228	0 14,546	14,625 24,753	0 14,161	12,069 23,058	13,901 19,383
M				16,280 27,143	0 14,518	13,313 25,262	13,581 21,895	
N					22,144 33,485	0 11,471	13,703 19,404	
O						14,064 20,819	BOC, 0 EFPD EOC, 335 EFPD	
P								
R

Burnup, MWd/mtU

Symmetric location of LTAs

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