ENTERGY

Entergy Operations, Inc.
River Bend Station
P.O. Box 220
St. Francisville, LA 70775

May 7, 1996

U.S. Nuclear Regulatory Commission
Document Control Desk
M/S P1-37
Washington, DC 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
Cycle 7 Startup Report

File No.: G9.5, G9.25.1.4

RBF1-96-0170
RBG-42888

Gentlemen:

In accordance with River Bend Station (RBS) Technical Requirements Manual TR 5.6.8, enclosed is a Startup Report that provides a summary of the startup physics testing conducted on the Cycle 7 core reload.

If you have any questions, please contact Mike Brandon at (504) 381-4506.

Sincerely,

FOR Rick J. King
Director - Nuclear Safety and
Regulatory Affairs

RJK/MKB
enclosure

IE26
1/1

cc: NRC Resident Inspector
P. O. Box 1051
St. Francisville, LA 70775

U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

Department of Environmental Quality
Radiation Protection Division
P. O. Box 82135
Baton Rouge, LA 70884-2135
Attn: Administrator

Mr. David L. Wigginton
U.S. Nuclear Regulatory Commission
M/S OWFN 13-H-15
Washington, DC 20555

ENCLOSURE

River Bend Station Unit 1

Cycle 7

Startup Physics Test Summary

OVERVIEW

River Bend Station (RBS) resumed commercial operation in Cycle 7 on February 13, 1996 following a Refueling/Maintenance Outage. The Cycle 7 reload consisted of replacing one GE7B and 232 GE8B General Electric fuel assemblies with 232 GE11 fuel assemblies and one GE8B re-insert fuel bundle. The following startup tests were performed during Refueling Outage (RF0) 6 and while attaining full power after RF06 and are summarized in this report.

1) Core Loading Verification

2) Control Rod Functional Testing

3) Shutdown Margin Determination

4) TIP Asymmetry

In addition to the above startup physics tests, the startup test program included: Core Monitoring System Verification and Recirculation System Calibration as well as other surveillance testing required by RBS Technical Specifications. The additional test results are available at the site on request.

CORE LOADING VERIFICATION

Purpose

Ensure each reactor fuel assembly is:

- in its correct core location
- oriented properly
- and seated properly in its support piece

Criteria

The reactor core is visually checked to verify conformance to the vendor supplied core loading pattern. Fuel assembly serial numbers, orientations, and core locations are recorded. A height check is performed to verify all assemblies are properly seated.

Results

The as-loaded core was verified for proper fuel assembly serial numbers, locations, orientation and seating in accordance with the RBS Cycle 7 core loading pattern. The core verification procedure was successfully completed on January 24, 1996.

CONTROL ROD FUNCTIONAL TESTING

Purpose

Verify operability of each control rod by:

- normal withdrawals and insertions
- ensuring it is latched to its control rod drive
- and moves at design speeds without excessive friction

Criteria

Functional testing of each control rod is performed to ensure proper operability. This testing includes withdrawal and insertion timing; coupling verification; friction testing, where required; and scram time testing.

Results

A control rod coupling check was performed in accordance with RBS Technical Specification Surveillance Requirement 3.1.3.5 each time a control rod was fully withdrawn.

Each individual control rod was timed during a normal withdrawal and insertion sequence. Control rods with stroke times outside the tolerance of normal stroke time +/- 20% were readjusted to within normal stroke time +/- 10%. This was in accordance with GE recommendations.

Three control rod drives were replaced and no fuel cells were disassembled during RF06. Friction testing was not required.

Each control rod was scram time tested during the Operational Hydro Test or reactor startup in accordance with RBS Technical Specification Surveillance Requirement 3.1.4.1 All of the control rod scram times were within the allowable limits.

One replaced Control Rod Drive Mechanism developed a flange leak during the vessel hydro. Following the hydro the flange O rings were replaced and this one control rod was scram time tested during reactor start-up in accordance with Technical Specification Surveillance Requirements 3.1.4.3 and 3.1.4.4 with satisfactory results.

SHUTDOWN MARGIN DETERMINATION

Purpose

To ensure:

- the reactor can be made subcritical from all operating conditions
- the reactivity transients associated with postulated accident conditions are controllable within acceptable limits
- the reactor will be maintained sufficiently subcritical to preclude inadvertent critically in the shutdown condition

Criteria

The subcritical demonstration verifies the reactor remains subcritical with the analytically determined strongest worth control rod fully withdrawn.

The in-sequence rod withdrawal shutdown margin calculation begins by withdrawing control rods in their standard sequence until criticality is achieved. The shutdown margin of the core is determined from calculations based on the critical rod pattern, the reactor period, and the moderator temperatures. To ensure no reactivity anomaly exists at criticality , the actual critical control rod positions are confirmed to be within 1% delta k/k of the predicted critical control rod positions. Reactivity Anomaly verification is performed by Technical Specification Surveillance Requirement 3.1.2.1 after reaching equilibrium xenon concentrations at greater than 80% reactor power.

Results

The subcritical demonstration was performed on January 24, 1996. The in-sequence critical shutdown margin surveillance procedure was completed on February 11, 1996.

The shutdown margin (SDM) at the beginning-of-cycle (BOC) was calculated to be 2.68 % delta k/k. The Cycle 7 "R" value is equal to 0.52% delta k/k, therefore, the Cycle 7 minimum shutdown margin at the most reactive point in the fuel cycle is 2.16 % delta k/k which is well within the RBS Technical Specification 3.1.1 requirement of 0.38% delta k/k.

The calculated reactivity differences between the actual and predicted SDM were checked at criticality and determined to be equal to a value of -0.23% delta k/k which was well within the Technical Specification Surveillance Requirement 3.1.2.1 limit of +/- 1% delta k/k. It was verified on February 19, 1996 that no reactivity anomaly was present by performance of Technical Specification Surveillance Requirement 3.1.2.1

TIP ASYMMETRY CHECK

Purpose

To determine the reproducibility of the Traversing Incore Probe (TIP) system readings

Criteria

An asymmetry determination is performed as part of a detailed statistical uncertainty evaluation of the TIP System. A complete set of TIP data is obtained at steady state conditions while greater than 50% rated core thermal power. A total TIP statistical evaluation is then performed by assessment of the random noise and geometric uncertainties. The results are evaluated to assure proper operation of the TIP System and symmetry of the core loading.

Results

The TIP reproducibility and symmetry uncertainty calculations were performed on April 5, 1996 at 100% core thermal power The total TIP uncertainty was evaluated to be equal to 3.4227 % which is less than the limit value of 6%.