UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-280/91-20 and 50-281/91-20

Licensee: Virginia Electric and Power Company
5000 Dominion Boulevard
Glen Allen, Virginia 23060

Docket Nos.: 50-280 and 50-281 License Nos.: DPR-32 and DPR-37

Facility Name: Surry 1 and 2

Inspection Conducted: July 8-12, 1991

Inspector: P. T. Burnett — Date Signed: 9-11-91

Approved by: R. V. Crlenjak, Chief, Operational Programs Section, Operations Branch, Division of Reactor Safety — Date Signed: 9/16/91

SUMMARY

Scope:

This routine, announced inspection addressed the areas of Unit 2, cycle 11, startup tests, thermal power analysis, power distribution monitoring, nuclear instrument calibrations, and followup of an unresolved item.

Results:

Following completion of the low power tests for Unit 2, cycle 11, it was determined that control rod F-6 (in control bank D) had been unlatched from its drive and fully inserted throughout the tests. Discussions were held with licensee personnel to determine and evaluate their conclusions that reperformance of the tests was not necessary following relatching of F-6. The licensee's conclusions were found to be acceptable (paragraph 2.c).

New feedwater flow venturis were installed in Unit 2 during the recent outage. Tests to compare performance of the new venturis with the steam flow venturis used for routine thermal power measurements were ongoing during this inspection. Review of the test results to date indicated that no single procedure was

capturing contemporaneously all of the data necessary to perform the analysis and obtain a cross calibration of the steam flow ventures. This issue was identified as an inspector followup item (paragraph 5).

Review of monthly core performance characteristics for Unit 1, cycle 11, confirmed that it has been operating with acceptable hot spot factors, reactivity balance, and power distribution throughout the cycle (paragraph 6).

The interval for surveillance of Unit 2 hot channel factors was found to be inviolation of Technical Specifications in one instance (paragraph 8).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

K. L. Basehore, Supervisor, Nuclear Engineering
W. R. Benthall, Supervisor, Licensing
*R. M. Berryman, Manager of Nuclear Analysis and Fuel
*R. E. Bilyeu, Licensing Engineer
D. Dziadosz, Supervisor of Core Design
*D. S. Hart, Supervisor, Quality Assurance
*J. W. Henderson, Lead Reactor Engineer
*M. R. Kansler, Station Manager
*D. C. Lawrence, Reactor Engineer
*M. A. Paul, Reactor Engineer
*J. A. Price, Assistant Station Manager
*E. R. Smith, Jr., Site Quality Assurance Manager
*T. B. Sowers, Superintendent of Engineering

Other licensee employees contacted included engineers, technicians, security force members, and office personnel.

NRC Resident Inspectors

M. Branch, Senior Resident Inspector
S. G. Tingen, Resident Inspector
*J. W. York, Resident Inspector

*Attended exit interview on July 12, 1991.

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Startup Tests for Unit 2, Cycle 11 (72700, 61708, 61710)

a. Precritical Tests

NPT-RX-007 (Revision 1), Hot Rod Drops, was performed at hot, full-flow conditions on June 2, 1991. All measured drop times satisfied the TS 3.12.C.1 requirement of being less that 2.4 seconds from interruption of power to dash pot entry. The timing measurements were made from individual chart records of rod drop speed versus time. The trace for rod F-6 was not as smooth, prior to deceleration into the dashpot, as that of the other rods. However, the test engineer's conclusion that the trace was the result of otherwise undetected electrical noise was not unreasonable; the drop time for F-6 was within the span of the other results.

Evaluation of the incore power distribution map obtained at

30 percent RTP (see paragraph 2.c) confirmed that control rod F-6 was fully inserted in the core (unlatched). The reactor was cooled down, the head removed, and rod F-6 latched to the rod extension.

Following a return of the system to hot, full-flow conditions, the procedure was performed again with acceptable results and traces for all control rods.

The inspector independently evaluated the rod drop time traces for several of the control rods for both sets of measurements. The inspector's results were consistently shorter than those reported by the licensee. Discussions with the test engineer produced agreement on the identifications of the point of power interruption and the point of dashpot entry on the chart records. However, the licensee used a ruler and the selected chart speed, 6 ins/sec, to determine time. The inspector counted the cycles of a 60Hz signal also present on the chart to determine time. The latter method is consistent with industry practice and guards against chart speed differing from nominal. In the current case, the chart speed appeared to be faster than nominal, which lead to a conservative estimate of control rod drop time. If the chart speed had been less than nominal, a more likely case with high speed recorders, the results would have been non-conservative. The licensee is considering revising the procedure, prior to the next startup test program, to use the imposed 60Hz signal to measure rod drop time.

b. Initial Criticality and Post-Criticality Tests

Initial criticality for Unit 2, cycle 11, was achieved using standard plant procedures, which provide for some monitoring of ICRR, but not to the extent usually seen at similar facilities for the first criticality in a cycle. No special effort was made to assure that the SRNIs were responding predominately and proportionately to neutrons. Many similar facilities use statistical reliability tests to assure confidence in the SRNIs prior to initial cycle criticality, during fuel movement in the vessel, and during lowered loop operations, when they are particularly vulnerable to a dilution accident. The licensee has a procedure for performing the statistical analysis, but has no procedure or practice for implementing it. The licensee is reviewing this issue.

Post-critical testing was guided by 2-PT-28.11 (Revision 1), Startup Physics Testing. Several good features of that procedure were noted:

At the beginning of the procedure, high flux trips and rod

stop setpoints were set to 85 and 81 percent of indicated power, respectively.

Both chambers of the PRNI connected to reactivity computer were confirmed to be supplying current to the computer.

The acceptable range of reactivity computer application was defined in terms of the measurement of its performance.

The acceptance criteria required that the ITCs measured during heatup and cooldown agree within 1 pcm/ Degrees F. However, that criterion was not repeated on the Test Result and Evaluation sheet.

Test Result and Evaluation sheets are prepared for most of the measured parameters, such as control rod worth and MTC. The sheets summarize the test results and conditions and compare the measured values with the predicted values at design conditions and at test conditions.

The acceptance criterion for the measured DBW worth was agreement within ±10 percent of the predicted value. This is a more stringent requirement than the ±15 percent imposed by ANSI/ANS- 19.6.1-1985.

Weakness noted in the procedure included:

The method for determining the point of adding heat was not well defined, and no margin between that point and the upper limit for zero power testing was specified.

-No Test Result and Evaluation sheets were provided for the point of adding heat, or for the incore-excore nuclear instrument correlation scheduled by the procedure.

-The source of data to correct the test control rod bank predicted worth for the position of the reference is not given.

One difference between the procedure and ANSI/ANS-19.6.1 was noted, but is not characterized as either a strength or a weakness: In the standard, error is defined as (P-M)/M where P is the predicted value of a parameter and M is the measured value. In the procedure, error is defined as (M-P)/P. At the limits, one approach can accept results rejected by the other and vice versa.

In practice, none of the results from this series of tests was near the limit of acceptability by either error analysis.

In the performance of the tests one particularly good

feature was noted: The temperature spans for ITC measurement for both heatup and cooldown met or slightly exceeded the 5 Degrees F specified by the procedure. The reliability of this endpoint-dependent measurement increases with increasing temperature span.

c. Testing with One Control Rod Inserted

The first flux map (at about 30 percent RTP) for Surry Unit 2, cycle 11, revealed that the control rod in position F-6 was fully inserted. That raised a question about the adequacy of the reactivity measurements made at zero power. All zero power tests (CBC at ARO, ITC/MTC, control rod bank worths, and DBW) had satisfied numerical acceptance criteria from predictive calculations, which, naturally, had not modelled a fully inserted rod.

The licensee then made calculations to predict the test parameters with F-6 inserted. Discussions with licensee NAF personnel confirmed that no shortcuts (interpolations, extrapolations, or perturbations) had been made to earlier calculations to make the new predictions. The new predictions were made using the same methods, with more effort, since core symmetry could not be assumed, as in the first predictions. As shown in Table 1, below, insertion of F-6 did not greatly change the predicted values of any of the test parameters, and the measured values satisfied the acceptance criteria for both sets of calculations.

The licensee's strength in core analysis was also demonstrated by accurate prediction of the power distribution at 30 percent RTP with F-6 inserted.

Since the purpose of the zero power tests is to assess or challenge the adequacy of the predictive methods, and all acceptance criteria for agreement between predictions and measurements were satisfied; the zero power tests are acceptable as performed.

The licensee's program for post-refueling startup test program is in substantial agreement with ANSI/ANS-19.6.1-1985, Reload Startup Physics Tests for Pressurized Water Reactors. No violations or deviations were identified.

3. Operation with Reconstituted Fuel (61702, 61706)

For purposes of thermal-hydraulic and critical heat flux analyses, a flow channel is usually defined or constructed from four fuel pins. The channel includes one-quarter of the cladding of each of the four pins plus the enveloped coolant. An unheated pin, such as a control rod or instrument thimble introduces a cold wall into the channel,

resulting in an equivalent heated diameter less than the hydraulic diameter. Cold walls have been demonstrated to reduce the critical heat flux of a flow channel.

Both units are currently operating with reconstituted fuel bundles, which, as a result of replacing damaged fuel pins with solid Zircaloy rods, have more cold wall in some flow channels than was considered in the development of the critical heat flux correlations used in the analyses of fuel performance. Much of the current concern for operation with excess cold wall is mitigated by the fact that all reconstituted fuel assemblies are once-burned, and, hence, are not peak power assemblies.

Westinghouse is scheduled to submit a topical report on this subject, for review by NRR, by mid-August 1991.

This issue will continue to be monitored in future inspections at this and similar facilities.

4. Nuclear Instrument Calibrations (61705)

A new single point method of correlating incore axial offset with the excore axial flux difference indicated by the power range nuclear instruments has been instituted at Surry. The statistical analysis and arguments for the single point method vice the earlier multipoint method appear to be well-founded. The licensee has identified those conditions under which return to the multipoint method will be required. The licensee's evaluations of this methodology are documented in the following reports, which were reviewed by the inspector:

- PM-325, Evaluation of Single Point Calibration Methodology for Surry Power Station, with four addenda, dated from September 14, to December 6, 1990.

- Technical Report NE-815, Evaluation of Excore Channel Single-Point Calibration Methodology, December 1990.

- NUCLEAR CORE DESIGN MANUAL, USER'S COPY, PART VII, CHAPTER E, Power Range Detector Calibration Versus Burnup, Revision 1, January 1991.

The licensee's implementation of this methodology is through procedures 1/2-NPT-RX-005 (Revision 0), Single Point Power Range Nuclear Instrument Calibration (Effective May 31, 1991). Review of completed copies of both procedures confirmed that the procedure was being performed with acceptable frequency and results.

No violations or deviations were identified.

5. Thermal Power Monitoring (61706)

New feedwater flow venturis were installed in Unit 2 during the past outage, and will be used to evaluate the performance of the steam venturis. (This licensee is one of the few to use steam flow venturis for plant calorimetrics.) Data are being collected by performance of the following procedures:

-2-ST-302 (Revision 0), Calibration of Steam and Feedwater Flow Transmitters at Power, and

-ENG-35.0 (Revision 0), Calculating Reactor Power, Delta T Setpoints and RCS Flow.

However, no one test procedure was capturing all of the information necessary to evaluate the flow and calorimetrics, and it was not clear that sufficient, contemporaneous data were being collected to accomplish those tasks. The licensee was reminded that industry experience is that new or cleaned feedwater venturis retain their precision for only 30 to 60 days of operation. The licensee acknowledged that only a limited time remained to complete the task. The licensee's activities in this area will be tracked as inspector followup item 50-281/91-20-01: Correlate thermal power and secondary side flow using both feedwater and steam flow venturis early in cycle 11.

No violations or deviations were identified.

6. Unit 1, Core Performance Surveillance Activities (61702, 61707)

The monthly Unit 1 Core Follow Reports for cycle 11 were reviewed by the inspector. The review confirmed that surveillances of F_Q, F_{dH}, and reactivity anomaly were being performed at the required frequency and with satisfactory results.

No violations or deviations were identified.

7. Audit of Reactor Engineering Activities (72700, 61702)

Much of the same subject area addressed in this report was audited by the licensee in Quality Assurance Audit 91-09, Nuclear Fuel. The inspector was provided a copy of the final audit report after the inspection, for in-office review.

With respect to onsite surveillance and test activities by reactor engineering and core design and follow activities by NAF, the report demonstrates that QA was able to perform a peer review of these activities and to identify the significant strengths and weaknesses of the audited organizations.

The resolutions of issues raised by the audit will be reviewed in later inspections.

8. Followup of Open Items (92701)

(Closed) Unresolved item 50-280 and 281/90-29-01: The interval between surveillances of hot channel factors exceeded 1.25 EFPM.

The Unit 2, cycle 10, records revealed that the interval between surveillances was 1.44 EFPM (44.8 EFPD) between July 18, 1990 and September 4, 1990. TS 4.10B requires that the hot channel factors of TS 3.12 shall be determined every EFPM. TS 4.02 allows a 25 percent tolerance on surveillance intervals, or a maximum of 1.25 EFPM, in this case. The licensee's position was that the language of the specification requires the surveillance in each full-power month, but does not limit the interval, which might then be nearly 60 EFPD. They further claimed that the NRC has found this interpretation and implementation of the surveillance requirement satisfactory in the past, but provided no documentation of that claim.

Review of the affected TS and their BASES was conducted in the regional office and by OGC and NRR/OTSB. The conclusion reached was that the surveillance should be conducted every EFPM (31 EFPD), with the grace period of TS 4.02 applying.

Hence, this issue has been identified as a violation, 50-281/91-20-01; The interval between surveillances of hot channel factors exceeded 1.25 EFPM resulting in a delay in assessing the acceptability of continued operation of the core.

9. Exit Interview

The inspection scope and findings were summarized on July 12, 1991, with those persons indicated in paragraph 1 above. The inspector described the areas inspected and discussed in detail the inspection findings. No dissenting comments were received from the licensee. Proprietary material was reviewed in the course of this inspection, but is not included in this report. The licensee was informed by a telephone call on August 16, 1991, that a violation would be issued for an excessive surveillance interval for

Unit 2 hot channel factors. During a later telephone conversation on September 4, 1991, licensee management stated they disagreed with the violation.

10. Acronyms and Initialisms used throughout this report

ANS	American Nuclear Society
ANSI	American National Standards Institute
ARO	All rods out
CBC	Critical boron concentration
DBW	Differential boron worth
EFPD	Effective full power day
EFPM	Effective full power month
F_{dH}	Enthalpy rise hot channel factor
F_Q	Heat flux hot channel factor
Hz	Hertz
HZP	Hot zero power
ICRR	Inverse countrate ratio
Ins/Sec	Inches per second
ITC	Isothermal temperature coefficient
MTC	Moderator temperature coefficient
NAF	Nuclear Analysis and Fuel Department
NRR	Office of Nuclear Reactor Regulation
OTSB	Technical Specifications Branch
OGC	Office of the General Counsel
pcm	Percent millirho
ppmB	Parts per million boron
PRNI	Power range nuclear instrument
QA	Quality assurance
RCS	Reactor coolant system
RTP	Rated thermal power
SRNI	Source range nuclear instruments
TS	Technical Specifications

TABLE 1

Surry 1, Cycle 11, Startup Physics Test Results

Parameter	Measured*	**	*	Predicted Tolerance
CBC (ARO,HZP), ppmB	1926	1930	1912	± 50
ITC (ARO,HZP), pcm/°F	-0.53	-1.04	-1.23	± 3.0
DBW (ARO,HZP), pcm/ppmB	-7.51	-7.31	-7.35	± 10%
Reference Rod Bank Worth, pcm	1375	1337	1368	± 10%
Total Worth of All Rod Banks, pcm	5480	5730	5623	± 10%

* With rod F-6 inserted
** Without rod F-6 inserted