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AmerGen

An Exelon/British Energy Company

Technical Specification 6.9.1.a.

1/23/03

United States Nuclear Regulatory Commission
Document Control Desk
Washington DC 20555

Subject: Oyster Creek Generating Station
Docket 50-219
Startup Report - Cycle 19

Enclosed with this cover letter is the approved Oyster Creek Generating Station Startup Report, for operating cycle 19. Please note that test requirement 28, "Rod Pattern Exchange", is currently scheduled for May 2003. A supplementary report will be submitted upon completion of the requisite testing.

If you should require any further information, please contact Mr. John Rogers, of my staff, at 609.971.4893

Very truly yours,

Ernest J. Harkness P.E., Vice President
Oyster Creek Generating Station

EJH/JJR

cc: Administrator, Region I
NRC Senior Project Manager
NRC Senior Resident Inspector

IE26

SUMMARY OF AMERGEN ENERGY COMPANY, LLC COMMITMENTS

The following table identifies commitments made in this document by AmerGen Energy Company, LLC. (Any other actions discussed in the submittal represent intended or planned actions by AmerGen Energy. They are described to the NRC for the NRC's information and are not regulatory commitments.)

COMMITMENT	COMMITTED DATE OR OUTAGE
Submit a supplemental report containing the results of Test Requirement 28.	Within 90 days of the completion of the testing.
And no other	None

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AmerGen
An Exelon/British Energy Company

OYSTER CREEK GENERATING STATION

CYCLE 19
STARTUP REPORT

SUBMITTED TO
THE U.S. NUCLEAR REGULATORY COMMISSION
PURSUANT TO
FACILITY OPERATING LICENSE DPR 16

JANUARY
2003

1.0 INTRODUCTION

Oyster Creek Technical Specifications Section 6.9.1.a requires that a summary report of plant startup and power escalation testing be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has different design or has been manufactured by a different fuel supplier, and modification that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications.

As a new fuel design GE-11(9x9) was introduced this cycle, a startup test report is required.

The 30 tests discussed in UFSAR Appendix 14.2A, and the 2 additional tests listed in UFSAR Table 14.2A-1, were reviewed to determine if they should be performed to support the fuel change. Each is discussed below, including a recommendation of whether or not to perform it, and the basis for that conclusion.

This report summarizes the plant startup and power ascension testing performed to ensure that no operating conditions or system characteristic changes occurred during the 18th refueling outage of Oyster Creek which could diminish the safe operation of the plant.

This is the first application of the GE11 product line at Oyster Creek. The GE11 fuel type has been approved for use by the NRC. GE11 fuel is mechanically,neutronically,and thermal-hydraulically compatible with the co-resident GE-9 fuel, RPV internals, spent fuel pool internals, refueling equipment, and other interfacing plant systems. ECR# 02-00789 justified and documented the technical acceptability of using GE11 fuel in Oyster Creek reactor. The potential impact on the systems and analyses has been analyzed: e.g.,loose parts, accident and transient analysis results, thermal limits, spent fuel pool cooling, fuel handling equipment, receipt inspection, Appendix R, noble metals, vessel fluence, etc. GE11 fuel complies with all required fuel design and licensing bases during steady-state, transient, and accident conditions.

Refueling and maintenance activities during the outage which may have any bearing on a fuel design change include:

- Core offload of 196 GE9 spent fuel bundles.
- Core reload of 190 new GE11 fuel bundles and 6 new GE9 fuel bundles.
- Replacement of 19 control rod drives.
- Installation of 2 *SLIMLINE* LPRMs
- Top guide inspections (Empty fuel from 4 cells)
- Disassemble/Reassemble entire Control Cell (46-43) for Bottom Head Inspection

Oyster Creek returned to service on 10-27-02 and reached steady-state full power for the first time in Cycle 19 on 10-29-02. Startup testing was completed on 11/08/02.

The successfully implemented startup test program ensures that the eighteenth refueling outage of Oyster Creek has resulted in no conditions or system characteristics that in any way diminish the safe operation of the plant.

2.0 UFSAR Described Tests

TEST 1 – Chemical and Radiological Sampling

Not Performed. This test serves to demonstrate that the chemical and effluent sampling equipment performs as designed. A fuel design change will not impact this equipment, so the test is not required.

TEST 2 – Control Rod Drives

This test is intended to demonstrate that the CRDs perform as designed. A fuel design change should not have any impact on this equipment. However, scram time data on all drives (required by Technical Specifications) is described below with the context that the half offset channel of GE11 does not result in unacceptable scram times.

Control Rod Scram Timing

All Control Rods were exercised to demonstrate normal notching capability prior to plant startup. Where HCU maintenance was performed, rods were notch timed and adjusted as necessary.

All control rods were scram time tested while reactor pressure was greater than 800 psig and were well within required Technical Specification limits as shown below. During the RPV hydro test, 42 rods were scram timed and the remaining 95 were scram tested prior to the plant increasing above 40% power. Additionally, there is no notable trend in scram speeds since last cycle. Therefore, there is no indication of system or configuration degradation.

Percent inserted (%)	Tech Spec Avg (sec)	Actual Avg (sec)	Maximum Individual Rod (sec)
5	0.375	0.293	0.340
20	0.900	0.699	0.810
50	2.00	1.523	1.750
90	5.0	2.595	2.950

TEST 3 – Fuel Loading

Fuel loading was performed in accordance with Oyster Creek Procedure 205.0 'Reactor Refueling' and NF procedures NF-AA-310 and NF-OC-300-1002. The objective was to load new fuel and shuffle the existing fuel safely and efficiently to the final loading pattern.

During fuel movement activities, all control rods remained fully inserted and at least two SRMs were operable, one in the quadrant where the core alteration was being performed and one in the adjacent quadrant (Technical Specification Section 3.9). Each fuel bundle remained neutronically coupled to an operable SRM at all times as verified by SHUFFLEWORKS. SRM count rates were recorded before and /or just after (yielding a before and after count) each core component move.

The final loading pattern includes 190 new GE11 fuel bundles, 6 new GE9B bundles, 184 once burned GE9B bundles and 180 twice burned GE9B bundles. The complete Cycle 19 core consists of all barrier fuel.

Core verification was completed on 10-18-02 in accordance with procedure NF-AA-330-1001. To ensure proper fuel loading into the core, the following steps were performed:

Proper fuel bundle serial number, location and orientation
Seating Verification
Debris inspection

The verified core loading map was compared with the Design Basis Loading Pattern (DBLP) and no discrepancies were found.

TEST 4 – <u>Shutdown Margin Testing</u>

Shutdown Margin Measurement test was performed by using the in-sequence critical method. Although the UFSAR discusses doing a local subcritical check (pull the highest worth rod and an adjacent rod to a point of specified SDM), an in-sequence test result will satisfy the requirement of the UFSAR test as OC Technical Specifications do not prescribe that a local demonstration or local subcriticality check be performed. Technical Specification 3.2.A and its Bases do not discuss the method of how the shutdown margin test is to be performed. Technical Specification Surveillance requirement 4.2.A also does not specify a method to be used.

Objective

The objective of the SDM measurement test is to demonstrate that the reactor will be subcritical throughout the fuel cycle with any single control rod fully withdrawn and all other rods fully inserted.

Description

Shutdown Margin was demonstrated with the "In-Sequence Critical" method. At criticality, correction factors were applied for moderator temperature, reactor period, worth of the "strongest" rod, the bias between local and distributed eigenvalue, and the "R" value for the cycle.

Acceptance Criteria

Technical Specifications require that the Core Keff not exceed a value of 0.9962 at any time during the cycle with the strongest operable control rod fully withdrawn and all other rods fully inserted i.e., the shutdown margin (SDM) is greater than or equal to 0.38% $\Delta K/K$. The BOC SDM test demonstrates that this requirement is met after core alterations have been made.

Results

Core shutdown margin was demonstrated by performing Procedure 1001.27 "Shutdown Margin Measurement Test" on 10/26/02. Control rods were withdrawn according to the startup sequence per Procedure 1001.4. SRM count rates were monitored during and after each control rod withdrawal. The reactor was declared critical at 1658 on 10-26-02 with RWM Group 3 Control Rod 10-31 at position 10, RWM step 5. Reactor water temperature was 178 degrees F. There were no inoperable control rods and the reactor period was 161 seconds.

Calculations

The BOC SDM value was calculated by subtracting the worth of the analytically determined strongest rod from the worth of all withdrawn rods and then applying the temperature, period, local versus distributed eigenvalue, and 'R' correction factors. This calculated SDM value from actual measurement was equal to 1.368% $\Delta K/K$. This value was verified to be greater than the required 0.38% $\Delta K/K$.

The difference between the predicted and actual SDM value is calculated as follows:

ΔSDM = SDM actual - (SDM predicted - R)
where R is the maximum decrease in SDM from BOC (which is 0 for this BOC start-up):
ΔSDM =(1.368 - (1.680-0)), or 0.250% $\Delta K/K$.

TEST 5 – **Radiation Measurements**

This test serves to demonstrate acceptable personnel dose levels. This test is meant for initial plant startup, and would not apply to a fuel design change.

TEST 6 – **Vibration Measurements**

This test is strictly an initial plant startup test to look at vibration characteristics of the internals and recirculation loops. The hydraulic characteristics of the fuel will not affect the vibration of guide tubes, the shroud, or the shroud head and separator assembly.

TEST 7 – **Control Rod Sequence**

This test is intended to demonstrate acceptable rod worths result from the sequence being used. This test predates the development of BPWS. The plant now uses a BPWS compliant sequence enforced by the Rod Worth Minimizer up to 10 % power as allowed by Tehnical Specifications. In sequence rod worths vary more as a function of the loading than the nuclear fuel type.

Criticality was achieved on 10/26/02 and actual critical eigenvalue was within 3 mk of the predicted critical eigenvalue (Procedure NF-AB-715).

Final Full power rod pattern was achieved on 11/17/02 following the completion of core spray surveillance. All thermal limits remained within their predicted values.

TEST 8 – **SRM Performance**

The test is written to demonstrate operability of the SRM instrumentation, and is not impacted by a fuel design change.

TEST 9 – **IRM Calibration**

The test is written as an initial plant startup test. A fuel change will not impact the calibration of the IRMs, and therefore it is not impacted by a fuel design change.

TEST 10 – **Reactor Vessel Temperatures**

This test is an initial plant startup test that determines temperature gradients in the vessel during startup and shutdown. Fuel design changes will not impact this.

TEST 11 – **System Expansion**

This test deals with thermal expansion of equipment and piping in the NSSS, and is not impacted by fuel design changes.

TEST 12 – **Main Steam Isolation Valves**

This test deals with leak tightness and stroke times of the MSIVs, and is not impacted by a fuel design change.

TEST 13 – **Isolation Condenser**

This test deals with operational characteristics of the isolation condensers, and is therefore not affected by a fuel change.

TEST 14 – **Recirculation Pump Trips**

This test determines the plant system response to a recirculation pump trip, including pump coastdown characteristics. A fuel change will not affect system response to the trip.

TEST 15 – **Flow Control**

This test is related to demonstrate the plant warranty capability on load following ability, which is unaffected by a fuel change.

TEST 16 – **Turbine Generator Startup**

This test checks operating characteristics of the turbine generator, which are unaffected by fuel changes.

TEST 17 – **Turbine Trip**

This test determines the plant response to a turbine trip, including feedwater and level control response. Fuel changes do not impact global system responses.

TEST 18 – **Generator Trip**

This test determines the plant response to a generator trip, including turbine overspeed and reactor pressure response. These responses are driven by system characteristics, and are not significantly affected by fuel changes.

TEST 19 – **Pressure Regulators**

This test determines the system response to a change in pressure regulator setpoint, and the ability of the backup regulator to take control. These responses are unaffected by fuel design changes.

TEST 20 – **Bypass Valves**

This test demonstrates the ability of the pressure regulator to minimize pressure disturbances due to a bypass valve actuating. This characteristic is not impacted by fuel changes.

TEST 21 – **Feedwater Pumps**

This is an initial plant test designed to demonstrate the dynamic response of the reactor to changes in the feedwater system, such as rapid level changes and feed pump trips and restarts. The dynamic response is driven primarily by system and equipment characteristics, rather than fuel characteristics.

TEST 22 – **Flux Response to Rods**

This test addresses the stability of the core with regard to the withdrawal of control rods. The dampening of radial instabilities due to rod withdrawal is driven by overall core geometry, power density, and max rod line, none of which are changing because of fuel design change.

TEST 23 – **LPRM Calibration**

Although the UFSAR describes LPRM calibrations at 25 %, 50%, and 100% power when no previous LPRM data is available, it was decided not to perform a full LPRM calibration using TIPS until 100% power. This was based on the fact that the original startup required LPRM calibrations at low powers as they had not been calibrated previously. For this startup, the LPRMs were still within their Technical Specification Calibration interval. A full set of TIPs was obtained at 75% CTP but LPRMs were NOT calibrated. TIPs were obtained so that a GAF file could be created. An LPRM calibration (the first with POWERPLEX III) was performed on 11/7 & 11/8 at 100 % power and equilibrium xenon conditions in accordance with Oyster Creek Procedures 1001.39 and 620.3.009. All LPRMs were calibrated except for three. One of the three required that the corresponding flux amplifier be re-strapped and the other two were downscales and bypassed.

TEST 24 – **APRM Calibration**

OC calibrates the APRMs to the Heat Balance. Manual Heat Balances were calculated at various power levels between 25 % and 100% and no anomalies were noted.

TEST 25 – **Core Performance Evaluation**

Test discusses determining thermal limits, bundle powers, core power, and core flow at various points in the power ascension. Throughout power ascension, Powerplex cases were manually triggered to provide current core conditions. No thermal limits were exceeded during these maneuvers.

TEST 26 – **Calibration of Rods**

The purpose of this test is obtain reference relationships between rod motion and reactor power in a standard sequence. During power ascension, POWERPLEX predictors were routinely performed prior to significant rod/flow maneuvers to provide the operators with the size of expected power change. No anomalies were noted.

TEST 27 – **Axial Power Distribution**

The following is a comparison of axial powers (off line(MCB) predicted versus on line adapted) at full power conditions. No anamolies were observed.

AXIAL NODE	PPLX KW/FT	MCB KW/FT	POWER DIFF %	K-INF
24	0.658	0.659	-0.325	0.8963
23	1.192	1.192	-0.325	0.9131
22	2.992	2.992	-0.325	1.0744
21	3.877	3.887	-0.006	1.0693
20	4.487	4.525	0.644	1.0512
19	4.777	4.846	1.309	1.0425
18	4.916	5.018	1.991	1.0343
17	4.903	5.036	2.689	1.0231
16	4.684	4.843	3.405	1.0193
15	4.676	4.847	3.7	1.017
14	4.708	4.872	3.561	1.0154
13	4.756	4.915	3.426	1.0142
12	4.822	4.976	3.294	1.0132
11	4.901	5.051	3.165	1.0125
10	4.986	5.131	3.04	1.012
9	5.101	5.201	2.108	1.0085
8	5.253	5.267	0.414	1.0053
7	5.467	5.392	-1.221	1.0055
6	5.675	5.506	-2.799	1.0061
5	5.797	5.535	-4.324	1.0083
4	5.693	5.351	-5.798	1.0165
3	5.107	4.761	-6.517	1.0331
2	3.867	3.604	-6.517	1.0754
1	1.046	0.974	-6.517	0.8421
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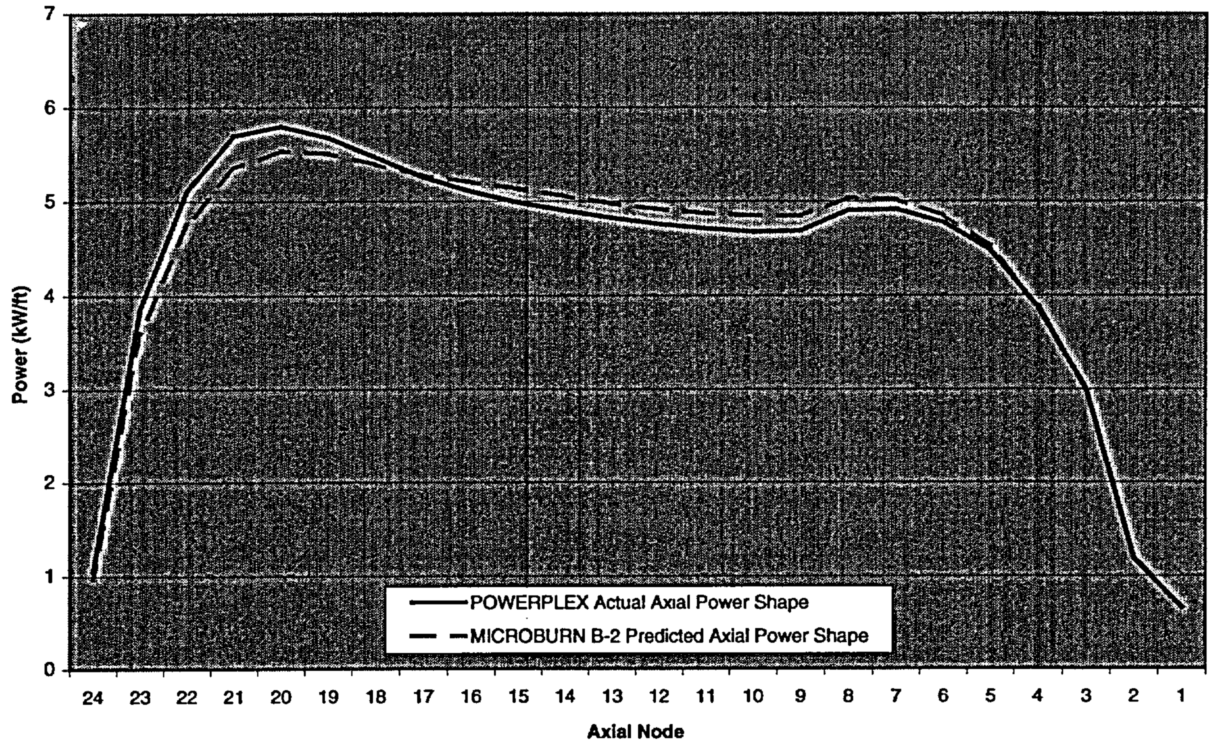
AVG	4.348	4.349	-0.080	1.0087

AXIAL POWER DISTRIBUTION



TEST 28 – **Rod Pattern Exchange**

The first Control Rod Sequence Exchange is scheduled at ~3300 MWD/ST (May, 2003). A supplementary report will be submitted following the completion of this exchange.

TEST 29 – **Steam Separator and Dryer**

This test deals with the carryover and carryunder performance of the separator/dryer, which is not affected by the fuel design.

TEST 30 – **Electrical Output and Heat Rate**

This test is related to demonstrating the plant warranty of net electrical output and heat rate. No impact due to the fuel change.

TEST 31 – **Loss of Auxiliary Power**

Plant response to a loss of auxiliary power event is a function of system design and equipment parameters, and is not significantly affected by fuel design changes.

TEST 32 – **LPRM Response**

During 1R19, two LPRM strings were re-installed in core locations 20-49 and 36-41. The "slimline" LPRM model were installed in these locations after 8 years of the core locations being flanged off/vacant. The "slimline" model is the same as a NA-300 except that the gland is machined to a smaller diameter. LPRM response on the newly installed LPRMs was verified per procedure NF-AB-719. LPRM 36-41 was verified on 10/27/02 and LPRM 20-49 was verified on 10/28/02.