NIAGARA MOHAWK POWER CORPORATION

NIAGARA MOHAWK

300 ERIE BOULEVARD WEST
SYRACUSE, N.Y. 13202

Regulatory File Cy.

December 15, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Morris:

Re: Provisional Operating License DPR-17
Docket No. 50-220

In our petition for escalation of licensed power to 1850 MW(t) from 1538 MW(t) for Nine Mile Point Nuclear Station, Unit No. 1 reactor, we described certain tests that would be performed when that power level was reached. That level was achieved and testing performed during the week of December 5, 1971. These tests, both static and dynamic in nature, correspond to the listing in Technical Supplement to Petition to Increase Power, Section II, page 33 and demonstrate reactor response and analysis of core operating parameters for several types of perturbations imposed in the system.

1. LPRM Calibration

The LPRM instrumentation was calibrated to read the average heat flux of the four corner rods surrounding each detector at the elevation of the detector. Individual TIP traces were used to calculate required power-distribution information. The correct LPRM reading for the existing reactor conditions were determined and the LPRM amplifiers adjusted accordingly.

2. Core Performance Evaluation

The principal core parameters of peak heat flux and Minimum Critical Heat Flux Ratio (MCHFR) were determined for several radial locations. The peak heat flux was found to occur in fuel adjoining LPRM location 04-25, at a value of approximately 94 watts/cm^2. The total peaking factor calculated was approximately 2.3, well below the allowable of 3.08. In addition, the Core Minimum Critical Heat Flux Ratio was found to occur at the location of the peak heat flux. This was approximately two feet above core bottom with a value of 3.33. This is above the minimum allowable MCHFR of 1.9 for the full recirculation flow condition.

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3. APRM Calibration

The Average Power Range Monitors (APRM) were calibrated several times during the testing period. The core thermal power was calculated from detailed heat balances of the reactor and associated systems. Following past practices, APRM's were adjusted to read 100% with core thermal power between 1538 MW(t) and 1850 MW(t).

4. Power Calibration of Rods

During the power ascension to the full design rating, various plant data was collected for the withdrawal of several control rod notches. Changes in the measured parameters were as expected.

5. Flux Response to Rods

Control rod 10-31 was withdrawn from position 18 to 20 and the response of LPRM 12-33C was recorded. The local neutron flux initially decreased as the rod was inserted to unlatch the collect fingers and then increased as the rod was withdrawn. The response of neutron flux, reactor water level and reactor pressure was well-damped with little or no overshoot from their final value at equilibrium.

6. Pressure Regulator Test

The performance of the pressure regulation system was demonstrated using both the Electrical Pressure Regulator (EPR) and the Mechanical Pressure Regulator (MPR) as follows:

a. nominal plus/minus 10 psi set point changes with the turbine control valves providing pressure regulation.

b. nominal plus/minus 10 psi set point changes with the turbine by-pass valves providing pressure regulation.

c. demonstration of pressure regulator takeover by both the EPR and MPR.

For all tests, the response of steam flow, reactor pressure, reactor power and reactor water level was well damped showing no significant oscillatory behavior.

7. By-pass Valve Test

The by-pass valve test consisted of opening and closing one by-pass valve to demonstrate the capability of the pressure regulator to minimize reactor pressure disturbances caused by changes in reactor steam flow. By-pass valve #12 I was opened in approximately 11.5 seconds using the by-pass valve functional test switch. The valve remained opened for approximately 65 seconds and then closed. The response of principal parameters was well-damped and exhibited no major oscillatory behavior.

8. Feedwater Pump Test

Reactor water level was first decreased then increased by approximately 6 inch increments with successive adjustments to the feedwater controller. Reactor power response in both cases was strongly damped during the subcooling changes produced by the level setpoint changes. In addition, the feedwater control system was very stable as the feedwater, vessel level response was well-damped.

9. Recirculation Flow Control

The master recirculation speed controller was adjusted to give plus/minus 10% flow changes. During the core flow reduction, core flow rate decreased at an average rate of 6% per minute. Reactor power and pressure decreased with little or no over shoot prior to settling at their respective final equilibrium value. During core flow increases, core flow rate increased at a faster rate but the system response again well-damped with little or no over shoot.

Response of the various systems and reactor are at least as good as anticipated and project the ability of the plant to handle transients of greater magnitude. Therefore, to minimize the detrimental effect of major transient testing on the fuel, we will defer the following tests until just prior to the next time the vessel head is removed, April 1972.

1. Maximum Recirculation Flow Changes
2. Five Recirculation Pump Trip
3. Turbine Trip

This is being done in the spirit of and conformity to recent changes in 10 CFR 20, whereby, we should hold off-gas releases to as low a practical value while at the same time assuring the public a dependable source of power.

During the interim period, auto-start recorders will be installed to obtain data necessary to prove-out the above tests should an inadvertent trip of the turbine take place.

Comprehensive reports of the test performed during the week of December 5, 1971 are in preparation and will be available for review.

Very truly yours,

F. J. Schneider

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Vice President - Operations