U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-244/84-26

Docket No. 50-244

License No. DPR-18 Priority Category C

Licensee: Rochester Gas and Electric Co.
49 East Avenue
Rochester, New York

Facility Name: R. E. Ginna

Inspection At: Ontario, New York

Inspection Conducted: December 3 - 7, 1984

Inspectors: C Petrone 1/23/85
C. Petrone, Lead Reactor Engineer date

C Petrone for 1/23/85
P. Wen, Reactor Engineer date

Approved by: LH Bettenhausen 1/23/85
L. H. Bettenhausen, Chief, TPS date

Inspection Summary:

Areas Inspected: Routine, announced inspection of the Cycle 14 startup physics tests including: core thermal power calculations, core power distribution, and follow-up on previous inspection findings. The inspection involved 26 hours onsite by one region-based inspector and 4 hours in the Region I office by another inspector .

Results: In the areas inspected, no items of non-compliance were identified.

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DETAILS

1. Persons Contacted

*T. Meyer, Technical Manager
K. Nassauer, QC Inspection Supervisor
*C. Peck, Nuclear Assurance Manager
*B. Snow, Plant Superintendent
S. Spector, Asst. Plant Superintendent
*J. Widay, Reactor Engineer
*W. Stiewe, QC Engineer

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*W. Cook, Resident Inspector

The inspector also contacted other licensee employees in the course of the inspection.

*Denotes those present at the exit meeting on December 7, 1984.

2. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (244/84-15-01) Control Rod Worths.
The results of control rod worth measurements for Cycle 14 startup physics testing indicated that the total measured rod worth for the three banks D, C, and B satisfied the acceptance criteria of ± 10%. However, the control bank B rod worth did not meet the individual bank rod worth acceptance criteria of ± 15%. Since the required shutdown margin (SDM) for the plant operation is closely related to the measured rod worth, the licensee's engineering analysis performed right after the startup test indicated that adequate SDM existed at the beginning of cycle (BOC). This conclusion was confirmed by the Inspector's independent calculation and was documented in Inspection Report 50-244/84-15. At that time, it was stated that recommendations for operation toward end of cycle (EOC), where excess SDM is at a minimum, awaited further analysis by the fuel vendor (Westinghouse) and a comparison of the observed cycle operation with predictions.

Subsequently, the rod worth calculation was re-examined by Westinghouse Nuclear Fuel Group and found to be correct. Although the startup physics testing program did not specify measurement of critical boron concentration for the control banks D, C, B inserted configuration, the boron sample data acquired during control bank B rod worth measurement did show that the measured critical boron concentration of 1005 ppm was in good agreement with the predicted value of 1010 ppm. In view of the good agreement between measured and predicted values for control bank D and C rod worth, this measured critical boron concentration for D+C+B in configuration indirectly confirmed the Control Bank B rod worth. The discrepancy observed during the startup physics test was attributed to the inherent error associated with the reactivity computer, especially when measuring the inboard bank rod worth.

The inspector further examined the reactivity anomalies plot performed since the beginning of this cycle. The inspector noted that the measured values are in excellent agreement with the predicted values. Based on the review of licensees' engineering analysis and close agreement of reactivity anomalies plot, the inspector concurred with the licensee's conclusion that the SDM at EOC is satisfied and no rod insertion penalty is required. This item is closed.

3.0 Core Thermal Power Calculation

3.1 Background: On October 25, 1984 the licensee notified NRC Region I that they had identified an error in the calculations used to determine the core thermal power at R. E. Ginna. The term used in the calorimetric equation for non-reactor heat inputs was in error by approximately 0.2% of the licensee power level of 1520 MWth (megawatts thermal).

The non-reactor heat input term is used during calculation of the core thermal power to correct for the reactor coolant pumps, charging, letdown, seal injection, seal return, reactor vessel support cooling, pressurizer heaters and ambient losses. The 1969 initial plant heat rate test established that the net heat input to the reactor coolant system from non-reactor heat was a positive 6.51 MWth which was 0.50127% of the 1300 MW licensed limit. For conservatism 0.5% was used in subsequent core thermal power calculations.

On March 1, 1972, the licensed thermal power limit was increased to the present 1520 MWth. However, the value used for non-reactor heat input remained at 0.5%. Since the change in thermal power rating did not significantly affect the non-reactor heat input, the value of 0.5% should have been reduced to a lower percentage. In September, 1984, the licensee reviewed these calculations using the values available from the 1977 Design Analysis and concluded that the calorimetric determination was in error by approximately 0.2% Core Thermal Power (CTP) in the non-conservative direction. The net heat input from the non-reactor sources was 17.574×10^6 BTU/hr which, with an additional 10% uncertainty factored in, resulted in a 0.3% of 1520 MWth correction factor. This was subsequently factored into the licensee's procedure O-6.3, Maximum Unit Power, which is used to calculate core thermal power.

3.2 Review During this inspection, the licensee's method for performing core thermal power calculation was reviewed and the effect of the error was evaluated by the inspector.

The licensee calculates the core thermal power each 8-hour shift in accordance with procedure O-6.3, Maximum Unit Power. The inspector reviewed the procedure and noted that its stated purpose was to assure that the licensed reactor power of 1520 MWth, averaged over any eight hour shift, was not exceeded. It requires that the power be reduced and the calculation be re-performed within two hours if the power is above 100%. The gain of the power range nuclear instruments must also be adjusted to agree with the calorimetric calculation.

The method of calculation used is based on a heat balance across the secondary side of the steam generators with corrections for non-reactor heat input and steam generator blowdown flow.

The inspector determined that the method used was appropriate and the corrected value for non-reactor heat input and blowdown flow were now correct. The inspector also reviewed the licensee's error analysis for calculation of blowdown flow and noted that the method of calculation yields a value that is conservative.

The inspector also reviewed the Ginna Final Safety Analysis Report (FSAR) to determine if the basis assumed in the FSAR were affected by the calculation error. The FSAR, Chapter 14, states that for accident evaluation, the initial conditions were obtained by adding the maximum steady state errors to the rated values. The assumed accuracy of measurement is ± 2% of rated power, so the actual power is assumed to be 102% for purposes of the accident analysis. Table 14-1, Instrument Drift and Calorimetric Errors, Nuclear Overpower Trip Channel, summarizes the assumptions made which establish the high power trip point at 109%.

	Set Point & Error Allowances % of Rated Power	Estimated Instrument Error % of Rated Power
Nominal Set Point	109	--
Calorimetric Error	2	1.55
Axial Power Distribution Effects on Total Chamber Current	5	3
Instrument Channel Drift and Set Point Reproductibility	2	1
Maximum Overpower trip point assuming all individual errors are simultaneously in the most severe direction	118	

The calorimetric error assumed in the error analysis is 2%. Adding the recently identified 0.2% error to the estimated calorimetric error of 1.55% the result is 1.75% which is still less than the 2% assumed in the FSAR. For added conservatism, the licensee keeps the high range, high power level trip set at

108% rather than the 109% assumed in the FSAR. Therefore, it appears that the 0.2% error in the core thermal power calculation is within the assumptions made in the safety analysis.

The inspector reviewed the Cycle 14 operating history to determine if the reactor had been run at a power level above the 1520 MWth licensed limit when suitably averaged. The licensee's reactor engineer stated that the plant is normally run between 99.5% and 100%. The inspector reviewed the Daily Reactor Power Logs for the period of August 28 through September 19, 1984 (the 0.2% error was found on September 20, 1984) and noted that average reactor power during that period was 99.54% (1513 MWth). The highest daily average power for that period was 99.8% (1517 MWth). From the start of the cycle 14 run on May 12, 1984 the daily average power level never exceeded 1517 MWth. During this review, the inspector also verified that anytime the shift calorimetric calculation indicated a power level above 100%, the power was reduced and another calorimetric was performed within two hours as required by Procedure O-6.3. Due to the licensees' policy of running the plant at slightly less than the licensed power limit, it appears that the average power level never exceeded 100% power limit during the time before the 0.2% error was identified.

3.3 Control Room Observation and Independent Calculation

On December 5, 1984 the inspector observed the performance of the shift calorimetric calculation in accordance with procedure O-6.3, Maximum Unit Power. The inspector discussed the procedure with the shift operations personnel who were knowledgeable and familiar with the procedure. The inspector verified that the instruments used for the feedwater venturi differential pressure, the blowdown flowmeters, and blowdown temperature detectors were in calibration. Since the calculated thermal power was 100.12%, the inspector verified that core power was reduced and the gain of the nuclear instrumentation adjusted as required by procedure 0.6-3. The inspector independently performed the calorimetric calculation and verified that the licensee's value of 100.12% was correct.

3.4 Crud Formation in Feedwater Venturi Nozzles

Another source of error in the core thermal power calculation is crud formation in the feedwater venturi nozzles used to measure feedwater flow. Westinghouse reports that this crud formation has been experienced at Ginna and ten other Westinghouse PWR's. The crud formation causes a reduction in throat area that results in a lower flow at a specified differential pressure. This causes the measured flow to be higher than the actual flow; consequently, the calculated core thermal power is higher than the actual core thermal power. Westinghouse reports that power losses have generally been in the range of two to three percent as a result of crud formation. This results in the plant's being operated conservatively by that amount. Since Ginna now routinely cleans these venturies during each refueling outage, the error is essentially zero at the beginning of each cycle and increases in the conservative direction during the cycle.

3.5 Moisture Carryon in Steam Generators

The licensee's present method of core thermal power calculation assumes that all the feedwater is converted to steam in the steam generators when, in fact, there is a small amount of moisture carryover. As a result, the actual core power is slightly less than the calculated core power by the amount of heat it would have taken to vaporize this carried-over moisture. In May, 1981, the licensee performed PT-20.2 "Steam Generator Moisture Carryover Measurement" which employed a radioactive tracer method to determine that the actual moisture carryover was .075%. Therefore, the calculated core thermal power is conservative by the energy equivalent of this very small amount of moisture carryover.

3.6 Summary: The 0.2% error in core thermal power calculation reported by the licensee did not affect the licensed core thermal power limit. The plant is normally run at slightly lower than the licensed power limit. The FSAR accident analysis assumes a calorimetric error of 2%, which is larger than the actual error (even with the additional 0.2% error). The plant hign power trips are set at 108% CTP, which is 1% more conservative than the 109% assumed in the FSAR, and required by Technical Specifications. The licensee does not take credit for the steam generator moisture carryover or crud formation in the feedwater venturies, both of which result in a more conservative calculation of core thermal power. The inspector had no further concerns in this area. Unresolved Item 84-22-04 is closed.

4.0 Quality Assurance (QA) and Quality Control (QC) Role in Startup Physics Testing

As discussed in Inspection Report 84-15, the licensee's QA/QC involvement in the Cycle 14 startup physics test program was found to be minimal. They committed to expand this involvement during the Cycle 15 startup.

During this inspection, the inspector questioned the Nuclear Assurance Manager and QC Engineer about their plans to increase QC involvement during the next startup. These plans included a revision to the startup physics test procedure to require that QC be notified prior to the performance of startup tests. The QC inspectors were given additional training which included observation of Cycle 14 startup tests and would include some classroom training. The inspector discussed this training with the QC foreman responsible for its implementation and reviewed the training schedule for the planned classroom training. The effectiveness of this training will be reviewed during the Cycle 15 startup physics tests.

The inspector also reviewed Audit Report 84-15, Cinna Station Refueling Activities, performed by the licensee's Quality Assurance department during the period from March 15 through May 25, 1984. The audit checklist included requirements to verify the accuracy of fuel accountability and inventory documents, review low power physics tests, and verify the qualifications of the startup physics test personnel. The auditor found them to be satisfactory. The inspector identified no additional concerns.

5.0 Core Power Distribution Limits

The procedures and methods used to verify plant operation within the power distribution limits, defined in Technical Specifications, were reviewed and discussed with cognizant licensee personnel. The flux data was obtained using the Movable Incore Detector System, then analyzed using the Westinghouse "Incore" computer code. Flux maps were performed during the startup test program at 23.6%, 46%, 71.5%, 83% and 100% core thermal power. During operation, they are performed monthly.

The inspector reviewed the results of the most recent flux maps taken at 100% power and noted that the control rod insertion, core power level, and burnup at the time of the flux map were part of the input to the code calculations. All incore dectors independently traversed a reference calibration tube. The power distribution limits, which are expressed as peaking factor, FQ and hot channel factor FΔH, were within the limits specified in the Technical Specifications and included an uncertainty correction.

The inspector also reviewed all of the Reactor Engineers' Incore Reduction Maps completed since Cycle XIV startup. These maps summarized the results of each flux map and verified their conformance to Technical Specifications. These results included the peak rod FΔH peak assembly FΔH peak core FQ; quadrant power tilt; axial offset; and the most limiting FQ. The inspector noted that the data reduction had been performed by the Reactor Engineer and reviewed and signed by the Technical Manager.

During inspection 84-15, the inspector noted that when the first core map was made at 25% CTP the results indicated a deviation between the predicted and measured FΔH's for some core locations. The licensee attributed this to the fact that only 13 of the 36 thimbles were ready for use when the flux map was taken. In addition, some thimbles were not in good alignment. During the present inspection, the inspector reviewed the results of the second flux map taken at 25% CTP, following the shutdown, and noted that all values were within predicted limits. No discrepancies were identified.

6.0 Target Axial Flux Difference

The inspector reviewed procedure 0-6.4.1, Reference Equilibrium Indicated Axial Flux Difference, and verified that it contained explicit instructions on how to determine the target axial flux difference at 100% power for each of the excore detectors. The procedure required that all four ΔI meters be operable, that equilibrium xenon conditions exist, and that the required rod configuration be established. The inspector verified that the target flux difference had been updated at least once each equivalent full power month, the values for target flux difference had been input into the process computer, and a plot of the target flux difference was provided to the reactor operators. No discrepancies were identified.

7.0 Control Room Observations and Facility Tours

The inspector observed control room operations for control room manning and facility operation in accordance with the administrative procedures and technical specification requirements.

No unacceptable conditions were identified.

8.0 Exit Meeting

The inspector discussed the inspection findings at an exit meeting on December 7, 1984.

No written material was provided to the licensee by the inspector at any time during this inspection.