SAFETY EVALUATION

GENERIC LETTER 83-28, ITEMS 3.1.1, 3.1.2, 3.2.1, 3.2.2, 4.1 AND 4.5.1

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 Introduction

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant failed to open upon an automatic reactor trip signal from the reactor protection system. This incident occurred during the plant startup, and the reactor was tripped manually by the operator about 30 seconds after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the undervoltage trip attachment. Prior to this incident, on February 22, 1983, at Unit 1 of the Salem Nuclear Power Plant, an automatic trip signal was generated based on steam generator low-low level during plant startup. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip.

Following these incidents, on February 28, 1983, the NRC Executive Director for Operations (EDO), directed the staff to investigate and report on the generic implications of these occurrences at Unit 1 of the Salem Nuclear Power Plant. The results of the staff's inquiry into the generic implications of the Salem unit incidents are reported in NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant." As a result of this investigation, the Director, Division of Licensing, Office of Nuclear Reactor Regulation requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System (RTS) Reliability Improvements. Within each of these areas various specific actions were delineated.

This safety evaluation (SE) addresses the following actions of Generic Letter 83-28:

-- 3.1.1 and 3.1.2, Post Maintenance Testing (Reactor Trip System Components)

-- 3.2.1 and 3.2.2, Post Maintenance Testing (All Other Safety-Related Components)

-- 4.1, Reactor Trip System Reliability (Vendor-Related Modifications)

-- 4.5.1, Reactor Trip System Reliability (System Functional Testing)

By letters dated November 4, 1983, Rochester Gas and Electric Corporation (RG&E or the licensee) described their planned or completed actions regarding the above items for R. E. Ginna Nuclear Power Plant.

2.0 Evaluation

2.1 General

Generic Letter 83-28 included various NRC staff positions regarding the specific actions to be taken by operating reactor licensees and operating license applicants. The Generic Letter 83-28 positions and discussions of licensee compliance regarding Actions 3.1.1, 3.1.2, 3.2.1, 3.2.2, 4.1 and 4.5.1 for TMI-1 are presented in the sections that follow.

2.2 Actions 3.1.1 and 3.1.2, Post-Maintenance Testing (Reactor Trip System Components)

Position

Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system (RTS) is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Licensees and applicants shall submit the results of their check of vendor and engineering recommendations (regarding safety-related components in the RTS) to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

Discussion

The licensee has completed a review of all maintenance procedures to assure that proper post-maintenance testing is being performed. This review was completed in June of 1983 and the necessary changes were incorporated as of November 1983.

The licensee performed additional review of administrative controls governing the preparation of maintenance procedures, emergency maintenance procedures, and system modification procedures. As a result

of this review, administrative controls have been strengthened to more adequately address post-maintenance testing of safety-related equipment, including reactor trip system components.

A comprehensive independent review of the Technical Specifications was performed to verify that all required surveillance testing is currently included in the surveillance program and is being tested at the required frequencies.

The licensee reviews vendor recommendations through their operational assessment program. This program is responsible for the review of operational activities at other power plants, in addition to performing the technical review of vendor recommendations. This has been an ongoing program for the licensee since 1980.

The licensee's engineering recommendations related to the topic of testing are generally in the form of test specifications. These specifications are converted to test procedures and receive a full PORC review before implementation. By these administrative procedures the license ensures that any appropriate test guidance is included in the applicable procedures.

Based on the above, the licensee has complied with the NRC staff position for Items 3.1.1 and 3.1.2 of Generic Letter 83-28.

2.3 Actions 3.2.1 and 3.2.2, Post-Maintenance Testing (All Other Safety Related Components)

Position

Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Licensees and applicants shall submit the results of their check of vendor and engineering recommendations (all other safety-related components) to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

Discussion

The licensee's single response for post-maintenance testing was inclusive for both Item 3.1, "Reactor Trip System Components" and Item 3.2, "All Other Safety-Related Components." Therefore, the

discussion for Items 3.2.1 and 3.2.2 is identical to the prior discussion for Items 3.1.1 and 3.1.2.

Based on the above, the licensee has complied with the NRC staff position for Items 3.2.1 and 3.2.2 of Generic Letter 83-28.

2.4 Action 4.1, Reactor Trip System Reliability (Vendor-Related Modifications)

Position

All vendor-recommended reactor trip breaker modifications shall be reviewed to verify that either: (1) each modification has, in fact, been implemented; or (2) a written evaluation of the technical reasons for not implementing a modification exists.

For example, the modifications recommended by Westinghouse in NCD-Elec-18 for the DB-50 breakers and a March 31, 1983, letter for the DS-416 breakers shall be implemented or a justification for not implementing shall be made available. Modifications not previously made shall be incorporated or a written evaluation shall be provided.

Discussion

The licensee reports that a review of Westinghouse correspondence concerning the Ginna DB-50 reactor trip breakers has identified two recommended modifications. The first was NCD-Elec-18, dated December 17, 1971, titled "Replacement of Undervoltage Attachments on Breakers in Reactor Trip Switchgear." This modificaiton was completed in 1972. The second modification was recommended in Westinghouse letter RG&E 83-596, dated May 9, 1983, which suggested that the reactor trip breakers be inspected for the presence of unused overcurrent trip brackets. These unused overcurrent trip brackets are to be removed from the breakers. The licensee completed the inspection and the recommended removal of the unused brackets by May 1983.

Based on the above, the licensee has complied with the NRC staff position for Item 4.1 of Generic Letter 83-28.

2.5 Action 4.5.1, Reactor Trip System Reliability (System Functional Testing)

Position

On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants. The diverse trip features to be tested include the breaker undervoltage and shunt trip features on Westinghouse, B&W and

CE plants; the circuitry used for power interruption with the silicon controlled rectifiers on B&W plants; and the scram pilot valve and backup scram valves (including all initiating circuitry) on GE plants.

Discussion

The licensee conducts independent functional testing of each of the two diverse tripping attachments. During this functional testing, breaker response times will be determined by deenergizing the undervoltage coil with the shunt trip coil blocked. The test will then be repeated with the undervoltage coil blocked (energized) and the trip coil activated (energized). The breaker clearing times will be recorded and trended for signs of degradation. The licensee conducts functional testing on an annual or refueling basis but states that more frequent (on-line) functional testing will be performed in the event that breaker response times increase and approach their maximum value. The licensee justification for not testing the reactor trip breakers during on-line conditions will be reviewed by the Office of Nuclear Reactor Regulation, on Item 4.5.3 of Generic Letter 83-28.

4.0 Conclusion

Based upon the foregoing discussions, the staff concludes that the licensee is in compliance with Actions 3.1.1, 3.1.2, 3.2.1, 3.2.2 and 4.1 of Generic Letter 83-28. Action requred by Item 4.5.1 will be evaluated by NRR in accordance with the requirements of Item 4.5.3 of Generic Letter 83-28.

Date: APR 10 1986

Principal Contributor: David M. Johnson, DRP, RI

APR 10 1980

DISTRIBUTION:
Docket Files
NRC PDR
Local PDR
M. Fairtile
G. Lear
P. Shuttleworth
OELD
L. Harmon
E. Jordan
B. Grimes
J. Partlow
T. Barnhart (4)
ACRS (10)
OPA
LFMB
PD#1 r/f
PD#1 s/f