SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 2

DOCKET NO. 50-410

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 due to a 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 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 - p 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)

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

By letters dated April 10, 1984, December 20, 1985, April 15, 1986 and March 18, 1987, Niagara Mohawk Power Corporation (NMPC - Licensee) described their planned and completed actions regarding the above items for Nine Mile Point Unit 2 (NMP-2).

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 and 4.5.1 for NMP-2 are presented in the sections that follow.

2.2 Actions 3.1.1 and 3.1.2, Post-Maintenance Testing (Reactor Trip System Components); and Actions 3.2.1 and 3.2.2, Post-Maintenance Testing (All Other Safety-Related Components)

Positions

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.

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 assure that any appropriate tests guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

Discussion

In a letter dated April 15, 1986, the licensee stated that the Administrative Procedure AP-2, "Production and Control of Procedure," requires review of test and maintenance procedures and Technical Specifications to assure that the post-maintenance operability testing of safety-related components including reactor trip system components is conducted. The review of test and maintenance procedures is effected by interdisciplinary review and cross disciplinary review. The review assures that the test procedures demonstrate that the equipment is capable of performing its intended safety functions prior to returning to service. The licensee also stated that all tests in maintenance procedures and Technical Specifications changes under go this review prior to implementation.

The licensee's departmental procedures S-IDP-PO, "Outline for I&C Procedures" and S-MI-GEN-002, "Maintenance Instructions for Writing Procedure," control the development of maintenance procedures. These two procedures require post-maintenance testing and are used by the reviewers to assure that appropriate post-maintenance testing has been incorporated.

The licensee's procedure AP-3.4.2 provides for the administrative control and evaluation of vendors information and recommendations. Accordingly, all NMP-2 related information recommendations from the reactor trip system supplier are reviewed and evaluated by Independent Safety Engineering Group (ISEG). In addition, NRC I&E Notices and Bulletins, INPO's Significant Event Reports and Significant Operating Experience Reports collectively provide a comprehensive and timely mechanism to assure that information pertaining to problems with safety-related equipment are identified and corrected. Also, through active participation in General Electric Operations Engineers Program the licensee has enhanced plant performance awareness, and has analyzed, evaluated and implemented General Electric recommendations as applicable to NMP-2. The program was designed to provide assistance in general plant operations and maintenance; provide assistance in interpretation of service information letters, backfits and other modifications; and increase flow and assimilation of information.

In letter dated March 18, 1987, the licensee stated that all procedures, required at present, for electrical maintenance, mechanical maintenance, and instrumentation and control maintenance have been issued and are in effect. These procedures were reviewed and approved in accordance with the licensee's Administrative Procedure AP-2. Currently, the licensee is in the process of replacing all Stone & Webster's project procedures by site service procedures and Niagara Mohawk departmental procedures, as applicable, and will complete this task by commercial operation.

In the foregoing letter, the licensee also stated that the latest revision of Administrative Procedure AP-2 as well as all site administrative procedures are now applicable to both Unit 1 and Unit 2. This met the licensee's commitment in the Unit 2 FSAR, Section 13.5.1.2, which called for incorporation of the Unit 2 into the existing site administrative procedures.

The licensee's present classification of safety-related components includes their subcomponents as well. NMP-2 utilizes the quality group classification system as delineated in the FSAR Section 3.2. The quality group classification applies to all NMP-2 structures, systems and components which are required to remain functional during and following a design basis event to insure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor and maintain it in a safe shutdown condition, and to prevent or mitigate the consequences of accident that could result in potential offsite radiation exposure.

The licensee's Engineering Assurance Procedure 3.1 describes review, control, and update of the procurement specifications of safety-related items. The procurement specifications include requirements for qualification testing, review, receipt and approval of testing documentation and vendor manuals. Maintenance and surveillance data extracted from the vendor documentation are transmitted to NMPC Project Engineering via Equipment Qualification Maintenance Program Data Sheet (EQMPDS). EQMPDS information is transferred to onsite maintenance management for incorporation into maintenance procedures in accordance with maintenance instruction MI-4.0.

The licensee had actively participated in the Nuclear Utility Task Action Committee (NUTAC) formed to control and utilize information regarding safety-related components. The exchange of information provides a mechanism for interchanges among utilities/vendors and utilities/regulator and established the Significant Event Evaluation and Information Network (SEE-IN) and Nuclear Plant Reliability Data System (NPRDS) programs managed by INPO. The licensee Procedure TDP-6, Nuclear Plant Reliability Data System Failure Reporting delineates NUTAC Vendor Equipment Technical Information Program (VETIP) to contribute information to the SEE-IN program via the NPRDS.

The licensee has stated that all corrective maintenance on safety-related equipment at NMP-2 are performed in accordance with Administrative Procedure AP-5.2, "Procedure for Repair" which specifies the requirements for post-maintenance testing (PMT) following any corrective maintenance. TDP-8, "Post-Maintenance Testing Criteria" provides guidance for the type of testing required based on the type of components and associated maintenance. Appendix C of AP-5.2 provides the pre- and post-maintenance testing criteria which establish the extent of testing following a maintenance activity.

The maintenance or repair works are initiated through the station Work Requests (WRs) in accordance with AP-5. The maintenance supervisor determines the availability and the adequacy of maintenance and test procedures to meet requirements of TDP-8. Upon completion of the work, the WR is returned to the control room for a review by the Station Shift Supervisor. Successful completion of the WR and required post-maintenance testing results in acceptance of the system or components for return to service by Operation.

The licensee also stated that all correspondence from Niagara Mohawk Project Engineering to the Nine Mile Point Unit 2 Station Superintendent were reviewed and cognizant personnel were interviewed to establish if any additional testing recommendations were still outstanding. No additional testing recommendations were identified.

Based on the above, the staff concludes that the licensee's actions are consistent with the NRC staff positions for Actions 3.1.1, 3.1.2, 3.2.1, and 3.2.2 of Generic Letter 83-28 and, therefore, acceptable.

2.3 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 NMP-2 reactor trip design features include a pair of dc solenoid operated backup scram valves. These valves are normally deenergized. At NMP-2 the scram pilot air system controls and supplies air to operate the scram valves and the scram discharge volume vent and drain valves through the two backup scram and two Redundant Reactivity Control System (RRCS) solenoid operated air valves. In an unlikely event, if the scram pilot valve fails to function, the action of the backup scram valves assures that the control rods insert, thus enhancing the reliability of the reactor trip function.

In a letter dated April 15, 1986, the licensee stated that current testing of the scram pilot valve is accomplished through the existing surveillance program. Accordingly, the trip system is functionally tested from the sensing instrument, through the trip logic circuitry, to the scram pilot valves. The surveillance procedures are written to test the one-out-of-two taken twice logic in such a manner that the channels are tested independently. This allows one-half of the necessary logic to "makeup," actuating the entire trip channel up to and including one out of the two scram pilot valves on every control rod's scram inlet and discharge valves in each channel.

In the proposed Plant Technical Specifications, the licensee indicated that the scram test will be performed each operating cycle to demonstrate operability and reliability of the system. The frequency of testing will be as follows:

1. For all control rods prior to thermal power exceeding 40% of rated thermal power following core alterations or after a reactor shutdown that is greater than 120 days;

2. For specifically affected individual control rods following maintenance or a modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods; and

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

In the above letter, the licensee indicated that the reactor trip system at NMP-2 is not designed for on-line testing of the backup scram valves. The current design would result in a full scram of one-half the control rods, if one of the backup scram valves was energized. Thus, functional testing of these valves during plant operation would require a plant scram, a significant challenge to plant safety systems and, therefore, a potential degradation of plant safety. The backup scram valves are nonsafety-related additions employed to enhance the reliability of the safety-related reactor trip system. Based o[illegible] redundancy of the backup scram valves and the scram pilot valve[illegible] licensee established that modifications to permit on-line testi[illegible] of the backup scram valves are not warranted. However, in response to Generic Letter 83-28, Action Item 4.5.2, the licensee indicated that the scram pilot valves are tested weekly during Average Power Range Monitor half scram test, and in accordance with the NRC guidance the backup scram valves will be tested during each refueling outage.

Based on the above, the staff concludes that the licensee's actions in this regard are consistent with the NRC staff position for Action 4.5.1 of Generic Letter 83-28 and, therefore, acceptable.

3.0 Conclusion

Based upon the foregoing discussions, the staff concludes that the licensee has complied with Actions 3.1.1, 3.1.2, 3.2.1, 3.2.2, and 4.5.1 of Generic Letter 83-28.

Dated:

Principal Contributor:

Madan Dev, Division of Reactor Safety, Region I

SALP INPUT

Facility: Nine Mile Point Unit 2

Docket Nos.: 50-410

TAC Nos.: None

Requested Date: July 2, 1985; R. W. Houston, Deputy Director

Functional Area: Adequacy of Licensee Submittal

COMPARISON OF PERFORMANCE TO CRITERIA ATTRIBUTES

Criterion 1: Management Involvement and Control in Assuring Quality

The licensee has reviewed, updated and revised the plant test and maintenance procedures and the Technical Specifications to implement the requirements of Salem ATWS Event follow-up. Accordingly, the post-maintenance functional testing of all safety-related equipment including Reactor Trip Breakers, is conducted in accordance with the approved procedures to demonstrate that the equipment is capable of performing its intended safety-function following repair and maintenance and prior to returning to service. Based on these observations, the licensee involvement and control in assuring quality is rated as Category 1 performance.

Criterion 2: Approach to Resolution of Technical Issues from a Safety Standpoint

The licensee has evaluated appropriate vendor and engineering recommendations regarding maintenance and testing of all safety-related equipment and incorporated them into the station test and maintenance procedures and the Technical Specifications. Based on the above, the licensee approach to resolve technical issues from a safety standpoint is considered Category 1 performance.

Criterion 3: Response to NRC Initiatives

The licensee, through an active participation in the INPO NUTAC Vendor Equipment Technical Information Program, maintains vendor information for all plant safety-related equipment and components current, complete and accurate. In addition, the licensee has extended the scope of review, evaluation and implementation of the Industry Event Review; installation procedure review; and control of the Operating and Maintenance Manual to enchance the program implementation. Based on the above, licensee response to NRC initiatives is rated as a Category 1 performance.

Conclusion:

The licensee submittal for response to Generic Letter 83-28, Salem ATWS event, was found to be adequately stated and enabled a clear understanding of the technical issues. The licensee's efforts to resolve staff questions concerning the issues were satisfactory.

Rating: Category 1

REFERENCES

AP-5.2	Procedure for Repair, Rev. 2
TDP-5	Administration of Operational Engineering Assessment Items, Rev. 1
TDP-6	Nuclear Plant Reliability Data System (NPRDS) Failure Reporting, Rev. 1
TDP-8	Post-Maintenance Testing Criteria, Rev. 0
TDP-9	Independent Safety Engineering Group, Rev. 0
N2-RAP-6	Post Reactor Scram Analysis and Evaluation, Rev. 0
NEL-014.G	Control and Distribution of Vendors Documents, Rev. 0
S-MI-GEN-002	Maintenance Instructions for Writing Procedures, Rev. 0
MI-4.0	Maintenance Instructions for Review and Implementation of Technical Requirements in Maintenance Procedures, Rev. 2
S-IDP-PO	Outline for I&C Procedures, Rev. 6
Engineering Assurance Procedure 3.1	Verification of Nuclear Power Plant Designs, Rev. 2

INPO Letter or NUTAC Recommended Enhancements