U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-333/90-19

Docket No. 50-333

License No. DPR-59

Licensee: Power Authority of the State of New York
P.O. Box 41
Scriba, New York

Facility Name: James A. Fitzpatrick Nuclear Power Plant

Inspection at: Scriba, New York

Inspection Conducted: June 25-29, 1990

Inspectors: A. L. DellaGreca 7/30/90
Neil Della Greca, Sr. Reactor Engineer, Plant Systems Section, EB, DRS — date

A. L. DellaGreca /RB 7/30/90
Ram Bhatia, Reactor Engineer, Plant Systems Section, EB, DRS — date

Approved by: CJ Anderson 8/1/90
Clifford Anderson, Chief, Plant Systems Section, Engineering Branch, DRS — date

Inspection Summary: Inspection of June 25-29, 1990 (Inspection Report No. 50-333/90-19)

Areas Inspected: Special, announced inspection to review the licensee's implementation of the post accident monitoring instrumentation in accordance with Regulatory Guide (RG) 1.97, Revision 2.

Results: Based upon this inspection, the inspectors determined that the licensee had implemented a program to meet the recommendations of RG 1.97. One Deviation pertaining to the use of a non-safety related transfer switch with the potential for paralleling of emergency buses was identified. The inspectors also determined that the licensee's corrective actions for one previously identified item were adequate for its closure.

No new violations were identified.

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DETAILS

1.0 Persons Contacted

1.1 New York Power Authority

* J. Erkan	Supervisor Project Engineer
* W. Fernandez	Plant Manager
* L. Guaquil	Director Project Engineering
* T. J. Herrmann	Systems Engineering Supervisor
* H. N. Keith	I&C Superintendent
* T. Landers	Superintendent Material Control
* R. Lesino	Superintendent of Power
* K. Moody	Senior Plant Engineer
* [illegible] Moskalyk	Plant Engineering Supervisor
* S. Mukherjee	Supervisor Engineer
* D. Ruddy	Plant Engineering Supervisor
* T. Savory	Sr. Project Engineer
S. Juravich	I&C Surveillance Coordinator
* D. Shih	Sr. Project Engineer
[illegible] Stranovsky	Nuclear O&M
J. C. Street	Sr. Electrical QA Engineer
* G. Tasick	Superintendent Quality Assurance

1.2 Nuclear Regulatory Commission

* W. Schmidt	Senior Resident Inspector

*Denotes personnel present at the exit meeting on June 29, 1990.

2.0 Background

The purpose of this inspection was to verify the licensee's implementation of instrumentation systems for assessing plant conditions during and following the course of an accident based upon the criteria specified in Regulatory Guide (RG) 1.97, Revision 2. The instrumentation systems were also inspected to determine if they were installed in accordance with Generic Letter No. 82-33, "Requirements for Emergency Response Capabilities" (Supplement 1 to NUREG-0737). This letter, issued on December 17, 1982, specifies those requirements regarding emergency response capabilities that have been approved by the NRC for implementation. The supplement also discusses the application of RG 1.97 to the emergency response facilities. This includes the control room (CR), the technical support center (TSC), and the emergency response facility (EOF) at nuclear power facilities. Regulatory Guide 1.97 identifies the plant variables to be measured and the instrumentation criteria for ensuring acceptable emergency response capabilities during and following the course of an accident.

Regulatory Guide 1.97 divides the Post-Accident instrumentation into three (3) categories and five (5) types. The 3 categories are noted as 1, 2, and 3. Category 1 has the most stringent requirements, whereas Category 3 the least stringent. The 5 types of instrumentation identified in the Regulatory Guide are types A, B, C, D, and E. Type A variables are plant specific and classified by the licensee; type B variables provide information to indicate that the plant safety functions are being accomplished; type C variables provide information on the breach of barriers for fission product release; type D variables indicate the operation of individual safety systems; and type E are those that indicate and determine the magnitude of the release of radioactive materials. Each variable type can be any category, except for type A which can only be category 1.

3.0 Scope

The NRC inspection scope included: identification of measured variables, method for measuring the parameter of interest (direct or indirect); display and recording methods used; redundancy of power supplies; independence and separation of electrical circuits; range and overlapping features of multiple instrument indicators; equipment qualification (Environmental and Seismic), equipment identification for RG 1.97 instruments; service, test and surveillance frequency.

4.0 Inspection Details

The inspectors held discussions with various members of the licensee's staff, reviewed drawings and procedures and selected variables for physical inspection. Walkdowns were performed for the sensing instruments at various locations of the reactor building and for display instruments in the control room to assess the implementation of RG 1.97.

The instrument variables reviewed were all Type A and included reactor water level, reactor pressure, torus water temperature, torus water level, drywell pressure, torus pressure, containment hydrogen and oxygen concentration, core spray system flow, and core spray system pressure. For each variable, the characteristics examined included the physical location of instrument component, function, physical and electrical separation, power sources, environmental and seismic qualification status, type and identification of display instruments, range and calibration.

An evaluation of applicable documents revealed that instruments located in a harsh environment were qualified for that environment and were included in the EQ master list. No master list existed for instrumentation located in a mild environment and no list existed for seismic Category I equipment. However, in the plant's Master Equipment List (MEL) each of the instruments within the scope of review was clearly identified to be environmentally qualified and classified as seismic Category I. In addition, each of the devices evaluated was included in the plant's Q list.

Instrument loops were found to be in calibration and calibrated within specified periods. Display instruments were found to be uniquely marked with a red mylar strip for ease of identification. Except as described below, scales and instrument ranges were found to be in accordance with the recommendations of RG 1.97. Recording was provided for at least one of the redundant instrument loops and, in most cases, by means of Class 1E recorders. An exception to this was the recording of the core spray system flow and pressure which was performed by the non-safety related plant EPIC computer.

Redundant instruments were found to be generally located in different areas of the plant with redundant display instrumentation in separate panels or enclosures. In one case, a Division I indicator (2-3LI-85A) and a Division II recorder (2-3LR-85B) were observed to be located side by side on the main control board. However, upon inspection, it was determined that separation was achieved using a metal enclosure around the indicator. The two cables were routed in different directions. In all cases, redundant instruments used redundant power supplies with adequate isolation between Class 1E and non Class 1E components.

4.1 Reactor Water Level

The licensee determined the reactor water level to be a type A variable. For this variable, RG 1.97 specifies that the range should be from the bottom of the core support plate to the lesser of the top of the vessel or the centerline of the main steam lines. For the James A. Fitzpatrick plant, this is 288" above the top of the active fuel. However, the safety related instruments provided by the licensee deviates by 63.5" on the upper end of the recommended span. The instruments only monitor reactor water level to 224.5". This deviation was originally justified by the licensee and reviewed by the NRC, as documented in the James A. Fitzpatrick Safety Evaluation Report (SER).

In addition to the above instruments, the control room instrumentation includes Category 3 level indicators which monitor the deficient range. The inspector determined that the combined ranges provide adequate coverage for this variable.

4.2 Computer Isolation

Where a Category 1 signals is used as input to a non-Category 1 system, RG 1.97 specifies the use of isolation devices which are fully qualified for use in Category 1 circuits. The inspector examined the circuits involved and determined that the isolation as well as the separation criteria had been adequately implemented by the licensee. In particular, the inspection revealed that digital and analog inputs to the plant computer were routed through Class 1E data acquisition subsystems (DAS) interconnected to the computer via fiber optic cables. However, the DAS cabinets themselves are fed by a non safety related uninterruptible power supply (UPS). This system was previously reviewed by the NRC for adverse impact on the safety related circuits and found to be acceptable.

Inputs to the the non safety related annunciator, for the variables evaluated, use Class 1E environmentally qualified Foxboro contact output isolators or Class 1E relays with coil to contact separation, as in the case of the hydrogen and oxygen analyzers. These devices were determined to provide acceptable isolation.

5.0 Findings

While reviewing documents relating to the subject inspection the NRC team discovered two items of concern which are described below.

5.1 Paralleling of Emergency Buses

Review of single line diagrams revealed that the non-safety related UPS described in section 4.2, above, is connected to two redundant 600 V emergency buses through a non-safety related, manually operated, transfer switch [illegible]1TS-7 and two Class 1E feeder breakers, one in each of the two load centers involved (L25 and L26). The review also revealed that no interlocks and no administrative procedures exist to prevent concurrent closure of both upstream feeder breakers. Therefore, a failure of the non-safety related transfer switch could parallel the redundant emergency buses. In this case, a fault anywhere on either bus, in the switch, or on any load supplied by either bus would result in a transient on both emergency buses at the same time.

The application constitutes a Deviation from the FSAR commitment to physical and electrical independence of redundant emergency power distribution systems (Section 8.5.6) in that the power transfer switch, if failed, could provide a path to interconnect the two emergency load centers, in the same manner as an automatic switching device would. Therefore, the switch does not meet single failure criteria, as specified in the same FSAR section. In addition, the switch was determined to be non-Class 1E.

Time-current curves were provided by the licensee which indicate that adequate coordination exists between the main (supply) and feeder (load) breakers of both load centers. However, no analysis was available to show that failure modes had been evaluated and, thus, show that redundant equipment would be able to perform their safety function.

Following the inspection, a further review of the one line diagrams available revealed that at least four pairs of redundant emergency motor control centers (C152 & C162, C251 & C261, C253 & C263, C254 and C264) supply power to loads through transfer switches and, therefore, can be similarly interconnected. One of the transfer switches (RWTS-01) is equipped with automatic throwover.

Resolution of the Deviation should consider credible failure modes affecting redundant equipment through transfer switches. In particular, the evaluation should address the effects of momentary voltage drops on AC circuits which rely on seal-in auxiliary relays and contactors for proper operation. (50-333/90-19-01)

5.2 Instrument Setpoints

Review of documents associated with reactor water level measurement revealed that the Technical Specification requirement for tripping the HPCI turbine and closing the RCIC steam line isolation valve on high level is 222.5" above the top of active fuel or 2" below the maximum span of the narrow range level instrumentation. The bistables involved are set to actuate at 221.2" on the basis of a memorandum from the NSSS supplier, dated March 11, 1985 and titled Fitzpatrick ATS Setpoint Determination. However, the analysis included in the memorandum uses instrument errors which are typical for normal, mild environments, e.g., 0.25% for Rosemount transmitters. Since qualification tests performed by Rosemount indicate potential errors in the order of 6-8% when the transmitters are exposed to LOCA environments, the licensee was asked if the 1.3" between the loop setpoint and the Technical Specification value was adequate margin under harsh environment conditions. The licensee suggested that the Technical Specification value already included a lot of margin. The licensee also informed the inspector that they had already recognized the need for formal calculations to address the effects of accident environments on loop accuracy and instrument setpoints and that they were in the process of developing a methodology to address them. The schedule for completing this effort, i.e., methodology and calculations, is currently set for December 31, 1990. Following the inspection on July 13, 1990, the licensee furnished a setpoint calculation for the loop in question. This calculation (FI-87-10S-SPEI), prepared to support a plant modification, adequately supports the setpoint selected. The inspector had no further questions regarding this matter.

6.0 Status of Previously Identified Items

(Closed) Violation (Item No. 50-333/87-14-02) involving the licensee's failure to provide documentary evidence that critical component characteristics had been met for commercial grade equipment used in safety related applications.

During the original review (Report No. 50-333/87-14), the inspector noted that Procedure No. EDP-17 provided the means for procuring commercial grade replacement parts for use in safety related applications. However, neither the procedure nor the checklist, which needed to be filled to evaluate applicability of commercial grade parts, imposed requirements to ensure that the commercial parts exhibited the critical characteristics necessary for their use in the safety related applications. In the followup inspection of July 1989, the NRC determined that the licensee had revised Procedure EDP-16 and prepared EDP-31 to address the dedication of commercial grade equipment. However, neither one of the procedures had undergone review and approval.

During the current inspection, the NRC inspector reviewed the status of the procedures and confirmed that adequate requirements were imposed for the procurement and dedication of commercial grade components used in safety related applications. The review revealed that EDP-16 and EDP-31 had been replaced by WACP-10.1.24 and WACP-10.1.25, respectively. To ensure that procedural requirement were being met, the inspector evaluated a small sample of recent procurements and technical reviews. The evaluation confirmed satisfactory utilization of procedures and adequate evaluation of parts for the dedication process. No safety concerns were observed.

This item is closed.

7.0 Exit Meeting

The inspectors met with the licensee representatives denoted in Section 1.0 of the report at the conclusion of the inspection, on June 29, 1990. At that time, the scope of the inspection and the inspection results were summarized. No written material was given to the licensee.