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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MARCH 2 5 1980

Docket No. 50-333

Mr. George T. Berry
General Manager and Chief
Engineer
Power Authority of the State
of New York
10 Columbus Circle
New York, New York 10019

Dear Mr. Berry:

Enclosed for your information is the Staff's evaluation for Fitzpatrick Nuclear Power Station of the actions you have taken to satisfy the Category "A" items of the NRC recommendations resulting from TMI-2 Lessons Learned. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on March 13, 1980. A list of meeting attendees is also enclosed.

Based on our review, we conclude that you have satisfactorily met all Category "A" requirements except for the installation of a valve position indicating system. We understand that this item will be completed on a schedule consistent with our Order dated January 2, 1980. The adequacy of certain implemented procedures and modifications will be verified by our Office of Inspection and Enforcement. Each of these is discussed in our evaluation.

Should you have any questions regarding our evaluation, please contact us.

Sincerely,

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Evaluation
2. Meeting Attendees

cc w/encls:
See next page

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EVALUATION OF LICENSEE'S COMPLIANCE WITH
CATEGORY "A" ITEMS OF NRC RECOMMENDATIONS
RESULTING FROM TMI-2 LESSONS LEARNED

Power Authority of the State of New York
Fitzpatrick Nuclear Power Plant

Docket No. 50-333

Date: March 25, 1980

I. INTRODUCTION

By letters dated October 22[1], November 21[2], 1979 and January 2[3], 1980 the Power Authority of the State of New York (licensee) submitted commitments and documentation of actions taken at the Fitzpatrick Nuclear Power Plant to implement our requirements resulting from TMI-2 Lessons Learned. To expedite our review of the licensee's actions, members of the staff visited the licensee's facility on March 13, 1980. This report is an evaluation of the licensee's efforts to implement each Category "A" item which was to have been completed by January 1980.

II. EVALUATION

Each of the Category "A" requirements applicable to BWRs is identified below. The staff's requirements are set forth in Reference 4; the acceptance criteria is documented in Reference 5. The numbered designation of each item is consistent with the identifications used in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY

The NRC requirement, as it is applicable to BWR's, is that provisions must be made such that the power-operated relief valves can be supplied emergency power when off-site power is not available. Further, for air-operated valves, emergency power must be available to the air compressors in order to provide a long term supply of air. The reactor water level instrumentation must also be capable of operating from emergency power.

The licensee stated at the site visit that all relief valves are supplied with redundant safety-grade control power. Long term operability can be obtained by use of nitrogen from the Containment Atmosphere Dilution (CAD) system. Item 2.1.5.a includes a description of this system. The reactor vessel water level instrumentation for safety system activation and control is powered by the on-site emergency power supplies.

Based on our review, we conclude that the licensee has satisfied the requirements of this item.

2.1.2 PERFORMANCE TESTING FOR BWR RELIEF AND SAFETY VALVES

The staff's position is that Boiling Water Reactor licensees shall functionally test the reactor coolant system relief and safety valves to demonstrate operability under expected operating and flow conditions. The Category "A" requirement is for the licensee to commit to perform an appropriate test program.

The licensee is a member of a GE BWR Owners Group and has committed[2] to a test program adopted by this Owners Group.[6]

We conclude that the licensee is in compliance with the performance testing requirements for relief and safety valves as outlined in NUREG-0578.

2.1.3.a <u>Direct indication of Power-Operated Relief Valves and Safety Valve Position for BWR's</u>

The staff's position is that BWR licensees shall provide a positive indication for reactor coolant system relief and safety valves. The valve position should be indicated and alarmed in the control room and derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe so that the operator is provided with an unambiguous indication of valve position. If the valve position indication is not safety grade, a reliable single channel direct indication powered from the emergency bus may be provided if backup methods of determining valve position are available. Further, the valve position indication should be seismically qualified consistent with the components or system to which it is attached. If seismic qualifications are not feasible by January 1, 1980, then justification should be provided and a schedule submitted for upgrading the system to meet the seismic requirements.

To meet the above position, the licensee has provided an acoustical system to monitor the position of each safety/relief valve. The acoustical system consists of a hermetically sealed piezoelectric sensor mounted on the downcomer piping of each safety/relief valve. The sensor is held in place by a special stainless steel band clamp and is connected to the preamplifier through the use of high temperature, low noise coaxial cable. The signal output from the preamplifier goes to individual Flow Detector Modules that provides position indication. The open, closed position for the safety/relief valves is also indicated on the main control panel. The preamplifiers are located inside the containment, where the temperature during an accident is higher than outside containment. Each Flow Detector module is located on an auxiliary rack of the main control room and provides indicator lights for "closed," "open" positions, and a "memory circuit." The "memory circuit" for each valve when activated stays on until manually reset; thus it provides an indication of valve actuation even though the valve may have since closed. If any of the flow detectors indicate a valve in the open position, a common dedicated single window of the plant annunciator is activated and its signal is also inputed to a 11 position indicating switch on the main control board. An on-line system test circuit for alarm has been provided.

The Fitzpatrick plant has 11 safety/relief valves. Each valve has the capability to be operated manually, however seven of these valves have been dedicated to the ADS function. The acoustic monitoring system installed on each safety/relief valves (located approximately 1-3 inches downstream of the valve discharge point) is not fully safety grade. The licensee has stated and we agree that the system is a reliable single channel system that provides direct indication and is powered from the emergency AC bus that has automatic transfer capability to a DC power supply. Also if both the AC and the DC power source fails to energize the bus, an automatic transfer to the redundant emergency AC power source will occur.

Back-up valve position indication information is provided and discussed in the emergency procedures so that the operator can make a diagnosis and take appropriate action. The back-up valve position indication is provided by temperature indicators. Each individual valve has a embedded type thermocouple attached to the tailpipe downstream of the valve discharge point. Signals derived from the embedded thermo-couple are readout on the process computer and a multi-pen recorder located on a auxiliary rack in back of the control room panel. A valve opening signal is alarmed and indicated on a dedicated window of the plant annunciator panel. The power for the back-up temperature monitor position indicators is provided from a non-Class 1E instrument bus. Therefore, in the event of a single failure, at least one position indicating system is available to provide the reactor operator with valve status.

The temperature indication instrumentation is already seismically and environmentally qualified and is available for backup verification of valve position. The acoustic monitoring system valve position indicators have not yet been seismically or environmentally qualified. The licensee stated that the position indication system and components, both inside and outside the containment is presently being environmentally and seismically qualified by participating in the B&W qualification program. The position indication system components will be seismically qualified in accordance with IEEE 344, 1975 and qualified for their appropriate environment in accordance with IEEE 323, 1974 by January 1981. This schedule is consistent with our requirements.

Based on our review of the licensee's submittal, we conclude that the licensee is in compliance with the direct indication of power-operated relief valves and safety valve position and schedule requirements for upgrading the system to meet the seismic requirements as outlined in NUREG-0578, and is, therefore, acceptable.

2.1.3.b Instrumentation for Inadequate Core Cooling

The NRC requirements, licensee actions and our evaluation thereof for this item will be evaluated separately by the NRC Bulletins and Orders Task Force and reported in NUREG-0645 which is incorporated herein by reference(7).

2.1.4 CONTAINMENT ISOLATION

The NRC requirements are that the licensee is to: (a) carefully reconsider the determination of which systems should be considered essential or non-essential for safety, (b) modify systems as may be necessary, to isolate all non-essential systems by automatic, diverse, safety-grade isolation signals, and (c) modify systems, as may be necessary, to assure that the resetting of the containment isolation signals does not cause the inadvertent re-opening of containment isolation valves.

The licensee's classification of essential systems was based on a determination of which are required or could be of direct aid in mitigating the consequences of an accident. All other systems which penetrate the primary containment are non-essential. The identification of each penetration, classification as essential or non-essential and the isolation signals for each is described in Table 3.7-1 of the Fitzpatrick Technical Specifications. The licensee's design of control switches for containment isolation uses spring-return-to-neutral and holding relays. For this type design resetting of the containment isolation signals does not result in the automatic reopening of valves. The one exception to this design is the system for sampling coolant water. In this case, the design consists of a set of series contacts that requires the affected isolation switches to be in the closed position before reset can be affected. Since this type design is limited to a single system (with only 2 valves involved) we find the licensee's overall design to be acceptable.

Based on the above we conclude that the licensee has adequately conformed to the requirements of this item.

2.1.5.a <u>Dedicated Penetrations for External Recombiner of Post-Accident External Purge System</u>

The staff's position is that licensees whose plant uses external recombiners or purge systems for post-accident control of combustible gas in the containment atmosphere should provide a containment isolation system tha is dedicated to that function only. The system's design should be edundant and meet our single failure requirements to that criterion 54 and 55 of the General Design Criteria are met and that the system is sized to satisfy the flow requirements of the recombiner or purge system. This requirement is applicable to those plants whose licensing basis includes requirements for external or purge systems for post-accident control of combustible gas in the primary containment.

The Fitzpatrick Station is designed to use a Containment Atmosphere Control (CAC) system prior to each startup and during routine operations to maintain the oxygen concentration in the primary containment atmosphere to less than 5 percent to ensure that combustion of the hydrogen and oxygen cannot occur. We have determined that the CAC system consists of the following major subsystems: The normal containment purge and exhaust subsystem, the containment inerting subsystem and the containment atmospheric make-up subsystem. These subsystems do not perform any safety function. Only those components associated with maintaining the containment isolation integrity (up to and including second containment isolation valve) are safety related and have been designed to seismic Category I requirements.

The Containment Atmospheric Dilution (CAD) system performs the safety function of limiting initial oxygen concentration to less than 5 percent in order to preclude a flammable mixture in the containment immediately following a LOCA and to maintain this inerted primary containment mixture on a long term basis following a LOCA. The CAD system is used during emergencies and as such has been designed to seismic Category I requirements; electrical components meet applicable portions of IEEE-279, and have suitable redundancy and interconnections so that a single failure of an active component will not render the system inoperable. The CAD system is functionally independent from the normal inerting system and its components include a storage vessel, electric vaporizers, redundant lines and valves and associated instrumentation. The nitrogen from the CAD system is injected into the drywell or torus using the purge air system lines. The CAD system branch lines are connected to the purge lines downstream of their redundant containment isolation valves. Two solenoid actuated isolation valves for each of the redundant torus and drywell CAD lines have remote control switches located in the main control room. In addition, two analyzers for hydrogen and oxygen have been provided for the containment drywell/torus that are redundant to each other and are designed to meet seismic and IEEE 279 requirements.

The licensee reviewed the CAD purge system at Fitzpatrick to verify that isolation provisions for piping and interconnected lines are single failure proof during CAD purge operation. The results of this review, reported in the January 2, 1980 submittal and affirmed at the NRR/OIE site visit verified that the containment isolation provisions for all lines except vent and purge lines are single failure proof. When these lines are in use, the outboard normal containment isolation purge valve must be opened to allow a flow path to the standby Gas Treatment System (SGTS). Thus complete reliance is placed on a single valve (the inboard containment isolation valve) to maintain the containment integrity. The licensee's proposed modification to rectify this deficiency is to:

(a) Change the 2" CAD purge line connections from up stream of the outboard containment isolation valve in the normal drywell and torus lines to downstream of the outboard valves.

(b) Add a new Motor Operated isolation valve to each of the 2" CAD purge lines so that it meets our single failure criterion.

(c) Revise the source of power to assure that valves are powered from different DC emergency buses so that it also meets our single failure criterion.

The licensee's schedule for completing this modification is prior to January 1981 which is consistent with our requirement. In the interim, the licensee has modified procedure F-SP-2 to caution operators of the potential single failure pathway and to identify isolation valves that could be operated to preclude venting the containment through the SGTS in the event of a failure of the inboard isolation valve.

Based on our review, we conclude that the licensee has met the requirements as outlined in NUREG-0578 and clarified by the letter of October 30, 1979 and therefore, is acceptable.

2.1.5.c Recombiner Procedures

The NRC requirements for this item apply only to those plants that include hydrogen recombiners as a design basis for licensing. We have determined that this item is not applicable to Fitzpatrick.

2.1.6.a Systems Integrity

The NRC objective is to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. The requirements are to implement practical measures to reduce leakage, report leakage measurements to the NRC and establish a preventive maintenance program to maintain leakage at as-low-as practicable levels.

Based on our review of licensee submittal and discussion with licensee during the site visit, we find that the licensee has tested and measured leak tightness of systems, developed leak reduction, and initiated a preventive maintenance program.

We conclude that the licensee has satisfied the requirements of this Category "A" item. The Office of Inspection and Enforcement (OIE) will review the licensee's procedures to verify adequacy. OIE will also verify the implementation of a preventative maintenance program and the completion of personnel training.

2.1.6.b Plant Shielding Review

The Category "A" requirements for this item are to perform a design review of current plant shielding to identify where corrective actions are needed to permit personnel access to vital areas, and to protect safety equipment.

The licensee has completed a general plant shielding review, identified vital areas for continuous and infrequent access and identified problem areas which will require additional shielding to comply with occupational dose criteria. The licensee stated that the radiation dose in the control room will be within required limits; however additional shielding may be necessary to reduce radiation levels in TSC and OSC. Emergency sampling stations will be provided outside of containment. A detail study to achieve an optimum solution to these problems is in progress. The licensee has reviewed plant shielding for degradation of safety system materials, such as TEFLON, which may require shielding or replacement.

Based on above, we conclude that the licensee has conducted a shielding review which satisfies the basic intent of this item. Completion of the shielding review, detailed description of necessary modifications, and plant radiation zone maps, will be submitted to NRC in a final shielding report. Our review of the final design modifications will be conducted as a Category "B" item.

We conclude that the licensee has satisfied the Category "A" requirements for this item.

2.7.7.a Auto Initiation of AFW

2.1.7.b AFW Flow

These items (2.1.7.a and 2.1.7.b) are unique to PWRs and are not applicable to the Fitzpatrick station.

2.1.8.A Post-Accident Sampling

The NRC objective is to quantify the degree of core damage in the course of an accident by radiological and chemical analysis of samples of reactor coolant and containment atmosphere. The Category "A" requirements are: (a) to review the design of reactor coolant and containment sampling system to determine the capability of personnel to obtain a sample (within 1 hour) under accident conditions without exposing an individual in excess of 3 Rem and 18 3/4 Rems to the whole body or extremities; (b) to review operational procedures of the radiological spectrum and chemical analysis facilities to determine the capability to quantify radioisotopes that are indicators of the degree of core damage; and (c) to describe proposed plant modifications.

The licensee has completed a design review of reactor coolant and containment atmosphere sampling systems. Additional shielding has been provided for the present systems, however, since personnel radiation exposure may exceed required limits, emergency sampling stations will be provided outside of the secondary containment. Samples can be collected, transported, and analyzed within 2 hours. Containment atmosphere samples can be taken from the present continuous H_2 and O_2 monitoring system. Emergency containment atmosphere sampling stations will be relocated outside of the containment.

Based on our review we conclude that the licensee has satisfied the requirement of this Category "A" item.

2.1.8.B High Range Radiation Monitors

The NRC objective is to have available adequate instrumentation to follow the course of the accident. The Category "A" requirements are to have procedures quantifying effluent releases in case existing instrumentation would go off scale ("provisional fix"). This includes a description of System/Method employed, and description of procedures for conducting all aspects of the measurement/analysis for noble gases, radioiodines, and particulate effluents.

The existing in-line effluent monitors are capable of detecting >1 Ci/sec of N.G., have continuous readout and alarm capability in the control room. Quantification of higher effluent releases is provided by portable gamma survey instruments, positioned on pre-selected location monitoring a small volume of the effluent sample line. This "provisional fix" has been installed and calibrated. Conversion factors have been developed for detecting effluents in excess of 10,000 Ci/sec. Since all station effluent releases are discharged via the stack, only one high range effluent monitor is required.

The licensee has committed to install two in-containment high range monitors (up to 10^8 Rad/hr).

Based on our review, we conclude that the licensee has satisfied the requirements of this Category "A" item.

2.1.8.C Improved Iodine Instrumentation

The NRC Category "A" requirements are that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where personnel may be present following an accident.

The licensee has available two car mou .ed air samples equipped with SCA, using charcoal cartridges (silver zeolite cartridges are also available for emergencies).

In addition the licensee has portable low volume air samplers, using charcoal cartridges, which can be purged and analyzed in the laboratory. The laboratory is located in the same building as the control room, TSC and OSC. The licensee stated that air samples can be obtained and analyzed within 10 minutes.

Based on our review, we conclude that the licensee has satisfied the requirements of this Category "A" item.

2.2.1.A Shift Supervisor Responsibility

The NRC requirement for this item is to revise, as necessary, the responsibilities of the Shift Supervisor such that he can provide direct, command oversight of operations and perform management review of ongoing operations that are important to safety and not be distracted from these important responsibilities by administrative details.

The licensee has revised Plant Procedure No. 1 to satisfy this requirement.

We conclude that the licensee has satisfied the requirements of Item 2.2.1.A to provide revised responsibilities and authority for the Shift Supervisor. Verification of the adequacy of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented by appropriate Inspection Reports.

2.2.1.b Shift Technical Advisor

The NRC requirement is for the licensee to provide an on-shift technical advisor (STA) to the shift supervisor to serve the two functions of accident assessment and operating experience assessment. As a supplement to the operating staff, the STA must be able to report to the control room within 10 minutes to assist in diagnosing an off-normal event.

The licensee has stated that he is using experienced nuclear engineers assigned on-shift at the site to perform the accident assessment. These engineers have received special training concerning off-normal operations. Plant Procedure No. 1 defines STA duties. The operating experience function is performed by a multi-disciplined on-site group of technical personnel.

We conclude that the licensee has satisfied the Category "A" requirements for this item.

2.2.1.c Shift and Relief Turnover Procedures

The NRC requirement is for the licensee to assure that procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift.

The licensee has revised Plant Procedure No. 4 to implement this item. We conclude that the licensee has satisfied the requirements of Item 2.2.1 to provide new procedures. Verification of the adequacy of the implemented checklists and logs will be performed by the Office of Inspection and Enforcement and will be documented by appropriate Inspection Reports.

2.2.2.A Control Room Access

The NRC requirement includes implementing procedures to limit access to the control room and establishing clear lines of authority in the control room in the event of an emergency.

The licensee has implemented procedures to satisfy the requirements for this item. Verification of the adequacy of the implemented procedures will be performed by the Office of Inspection and Enforcement and will be documented by appropriate Inspection Reports.

2.2.2.b Technical Support Center

The NRC requirement is that each licensee establish and maintain an onsite technical support center (TSC) separate from and in close proximity to the control room. The TSC should have reliable communication systems and plant as-built technical data to provide information to those individuals knowledgeable and responsible for engineering and management support to reactor operations in the event of an accident. Further, the licensee must describe the long range plan to upgrade the TSC to meet the Category "B" requirements.

The licensee has developed an interim onsite technical support center (TSC) and has designated that location in the emergency plan. During the NRR/OIE site visit we toured the TSC. Direct telephone communications and airborne and radiation monitors provide warning and monitoring capability. Direct access to all plant drawings and records is provided as part of the TSC. Revised emergency procedures are in effect directing the operation of the TSC.

We conclude that the licensee has satisfied the Category "A" requirements for this item.

2.2.2.c Operational Support Center

The NRC requirement is to establish an area in which shift personnel can report for further instructions from the operations staff.

The licensee has designated Onsite Operational Support Center separate from the control room. During the site visit the staff toured the designated Operational Support Center. The center has telephone communications. The licensee's Emergency Plan covers this Center.

We conclude that the licensee has satisfied the requirements of 2.2.2.c.

NRR ITEM: REACTOR COOLANT SYSTEM VENTING

As specifically related to BWRs, the Category A requirements of this item is to provide current design information to demonstrate that non-condensable gases can be vented from the primary coolant system.

The licensee's submittal dated October 22, 1979[1] provided design information for the Fitzpatrick Plant. The capability includes safety/relief valves which discharge to the suppression pool, a vessel head vent to the radwaste system, a vessel head vent to one of the main steam lines and steam driven HPCI and RCIC pumps which discharge to the suppression pool.

The design information is sufficient to satisfy the Category "A" requirements for this item.

References

1. Letter, PASNY (Early) to NRC (Eisenhut), October 22, 1979.
2. Letter, PASNY (Schnieder) to NRC (Ippolito), November 21, 1979.
3. Letter, PASNY (Early) to NRC (Ippolito), January 2, 1980.
4. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, September 13, 1979.
5. Letter, NRC (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, October 30, 1979.
6. Letter, BWR OWNERS GROUP (Keenan) to NRC (Eisenhut), December 14, 1979.
7. NUREG-0645 Report of the Bulletins and Orders Task Force, January 1980.
8. Letter, NRC (Denton) to IELP (Root) with enclosure: ORDER TO SHOW CAUSE, January 2, 1980.

PASNY (Fitzpatrick) LL Meeting 3/13/80

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