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John C. Brons
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December 24, 1985
JPN-85-91

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. Daniel R. Muller, Director
BWR Project Directorate No. 2
Division of BWR Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Emergency Response Capability - Conformance to Regulatory Guide 1.97 Revision 2

References:
1. NYPA letter, J. C. Brons to D. B. Vassallo, dated November 30, 1984 (JPN-84-77) regarding response to Generic Letter 82-33.
2. NYPA letter, J. C. Brons to D. B. Vassallo, dated June 28, 1985 (JPN-85-53) regarding revised schedule for implementation of Regulatory Guide 1.97.
3. NRC letter, D. B. Vassallo to J. C. Brons, dated November 5, 1985 transmitted EG&G preliminary Technical Evaluation Report regarding the same subject.

Dear Sir:

In Reference 1, the Authority responded to Section 6 of Generic Letter 82-33 and described plans and schedules for implementing Regulatory Guide 1.97 Revision 2 at the FitzPatrick Plant. Exceptions to certain of the guide's regulatory positions were justified in that letter. The implementation schedule was revised and updated in Reference 2.

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Reference 3 transmitted a preliminary Technical Evaluation Report (TER) documenting the NRC's review of the Authority's proposed plans. The TER identified five open items, requested a response and solicited the Authority's comments.

Attachment 1 replies to the five open items identified in the TER. Plans and schedules for two Regulatory Guide 1.97 variables, deferred until after the Authority had responded on the new rule on Anticipated Transients Without Scram (ATWS), are included as Attachment 2. Authority comments on the TER are included in Attachment 3. Attachment 3 also corrects the implementation schedule included with Reference 2.

If you have any questions regarding this information, please contact Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

John C. Brons

John C. Brons
Senior Vice President
Nuclear Generation

cc: Office of the Resident Inspector
U. S. Nuclear Regulatory Commission
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ATTACHMENT 1 TO JPN-85-91

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
DPR-59

NYPA Response to NRC letter of November 5, 1985
and the Technical Evalution Report regarding
conformance to Regulatory Guide 1.97 Revision 2

NRC Comment 1: Neutron flux -- the licensee's present instrumentation is acceptable on an interim basis until Category I instrumentation is developed and installed (Section 3.3.1).

NYPA Response: The Authority will continue to use the existing neutron flux instrumentation.

The Authority will follow the industry development of new neutron flux monitoring equipment. Replacement equipment will be evaluated when it becomes avaliable. The Authority will consider installing new neutron flux monitoring instrumentation at that time.

NRC Comment 2: Radiation exposure rate -- the licensee should show that the ranges supplied for this variable encompass the radiation level at the instrument location (Section 3.3.4).

NYPA Response: (See response to NRC Comment 5 below.)

NRC Comment 3: Standby liquid control system storage tank level -- environmental qualification should be addressed in accordance with 10 CFR 50.49 (Section 3.3.7).

NYPA Response: As the Authority stated in Reference 14, there are no components in the Standby Liquid Control (SLC) system which require qualification under the provisions of 10 CFR 50.49 (Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants).

The SLC system design basis assumes the need for an alternate method of reactivity control without a concurrent LOCA or high-energy line break. The environment in which the SLC tank level instrumentation must work is therefore a mild

environment for equipment qualification purposes.

Therefore, no changes to the existing SLC tank level instrumentation are necessary to comply with the requirements of 10 CFR 50.49.

NRC Comment 4: RHR service water flow -- the licensee should either show that the range is adequate or provide the recommended range (Section 3.3.8).

NYPA Response: An error was made in the Position Summary Table included with Reference 1. The table entry associated with RHR Service Water System Flow (Items A9 and D22-B) should have listed an installed instrument range of zero to one hundred and fifty percent (0% - 150%) of design flow. The installed instrument range (0% - 150%) exceeds the range required by the regulatory guide (0% - 110%). Therefore, no plant modification or futher justification is necessary.

NRC Comment 5: Reactor building or secondary containment area radiation -- the licensee should show that the ranges supplied for this variable encompass the radiation level at the instrument location (Section 3.3.10).

NYPA Response: The existing Area Radiation Monitoring System (ARMS) consists of thirty monitors located throughout the plant. Most instruments have a maximum indication of 1 R/hr. ARMs for the refueling floor and chemistry laboratory have ranges of 1000 R/hr and 0.1 R/hr, respectively. Alarm points range from 1 to 200 mR/hr, depending on detector location.

The Authority considers the existing Area Radiation Monitoring System to be adequate for the following reasons:

1. The Authority has previously stated that it is not necessary to enter the FitzPatrick Reactor Building following an accident (Reference 7).

   As a part of the our efforts to respond to NUREG-0578, Item II.1.6.B, and NUREG-0737 Item II.B.2, the Authority performed a study to predict radiation levels in the plant in the vicinity of systems that may, as the result of an accident, contain highly radioactive material. This report identified vital areas and equipment in which personnel occupancy may be unduly limited or safety-related equipment

unduly degraded during post-accident operation. This study used the conservative assumptions of a post-accident release of radioactivity equivalent to that described in Regulatory Guide 1.3 (i.e., the equivalent of fifty percent of the core halogens, one hundred percent of the core noble gases, and one percent of the core solid fission products are contained in the primary coolant). The study considered only systems outside primary containment which may contain primary coolant or gases: HPCI, LPCI, RHR, Core Spray, Containment Air Dilution (including torus/drywell purge and vent system), Standby Gas Treatment and reactor coolant sampling lines. The following areas were considered in this analysis: Control Room, Technical Support Center, Primary Coolant System Sampling Stations, motor control center locations, electrical control panel locations, manual isolation and control valve locations for vital equipment, Chemistry Laboratory and Counting Room, areas where emergency maintenance may be desirable, and other general access areas. Twenty detector points were modeled in the analysis.

The results of this study show that twenty-four hours after the accident, more than one-half of the detector points modeled would be exposed to dose rates less than 1 R/hr (i.e. on-scale for most of the installed area radiation monitors). After four days, Reactor Building doses rates were predicted to range from a maximum of 38 R/hr to zero R/hr (3.7 R/hr average). After fourteen days, the maximum dose rate decayed to less than 20 R/hr with an average of 1.7 R/hr.

The NRC's Office of Inspection and Enforcement conducted an inspection (Reference 9) in March of 1983 to review our post-accident shielding design review. As part of that inspection (Section 2e, Vital Area Accessibility Procedure Review), the inspectors reviewed three procedures that would be used post-LOCA. The inspectors determined that these three procedures (1) could be completed from the Control Room, (2) contain appropriate provisions to assure controlled access to vital areas for post-accident operations, and (3) post-accident doses to plant personnel would be within the guidelines of NUREG-0737. Overall,

our shielding design review was found to be consistent with the guidelines of NUREG-0737.

The use of the existing area radiation monitors in classifying emergencies is described in NYPA procedure IAP-2, "Classification of Emergency Conditions."

The source terms used in the post-accident shielding analysis are very conservative. Studies of the accident at TMI-2 confirm this (Reference 16).

While some of the existing area radiation monitors will read off-scale during the early phases of the accident and recovery, this is acceptable since entry into the Reactor Building is not required in the short or medium term.

2. An extended-range area monitoring system would not, by itself, provide adequate information during an emergency situation. Local radiological surveys and airborne contamination samples would be required prior to any entry into the Reactor Building regardless of whether or not extended-range area radiation monitors were installed.

3. Portable radiation survey instruments are adequate as a primary source of information considering that Reactor Building entry is not necessary in the short or medium term. These instruments are maintained and used as part of normal plant operations. Plant personnel are trained to use this equipment and are familiar with it.

4. In response to an NRC Emergency Preparedness Appraisal (Reference 10, Appendix B, Item 11), the Authority performed an engineering study on upgrading the existing Area Radiation Monitoring System. This study found an upgrade not to be cost effective.

   During a subsequent I&E inspection (Reference 11), an NRC inspector reviewed the study and it's conclusion. As a result, the inspector closed the open item.

5. The Authority does not consider area radiation monitors a feasible or desirable way of

detecting a breach of primary containment. Primary containment breaches are better detected by plant noble gas effluent monitors. The BWROG has adopted this same position (Reference 12).

In general, Reactor Building radiation exposure rate will reflect the levels in the drywell atmosphere and ECCS piping (reactor coolant). Area radiation readings could be distorted as a result of direct shine from piping and equipment carrying drywell atmosphere or reactor coolant. Local radiation exposure rate monitors could indicate ambiguous exposures due to the amount and arrangement of piping, the quantity of electrical penetrations, hatches and their scattered arrangement.

6. High-range radiation monitors have been installed in other parts of the plant in response to other NUREG-0737 items.

   The existing High-Range Effluent Monitori. System (HREMS) consists of three noble gas monitoring units. HREMS units are connected upstream of the Turbine Building exhaust sampler, Radwaste Building exhaust sampler, and Main Stack effluent monitor. Each monitor contains two redundant ion chamber detectors and with a range of 0.1 to 10,000 R/hr. Displays are mounted in the Main Control Room. In addition, three Control Room annuciators are associated with each HREMS monitoring unit - FAILURE (loss of power or circuit malfunction), ALERT (high radiation), and HIGH (high-high radiation). The ALERT and HIGH annuciators indicate when release levels begin to approach Site Emergency or General Emergency Conditions (as defined by the FitzPatrick Site Emergency Plan). Dose rates are also recorded on three, two-pen recorders located adjacent to the instanteous rate displays. (Futher information on the HREMS can be found in Section 11.4.5.1 of the updated FitzPatrick Final Safety Analysis Report.)

7. Existing process and effluent monitors will also provide indication of releases or breaches in systems used post-accident.

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
DPR-59

Plans and Schedules for the Implementation of Two Regulatory Guide 1.97 Revision 2 Variables - Standby Liquid Control Tank Level Instrumentation and Standby Liquid Control System Pump Flow

In the Authority's June 28, 1985 letter (Reference 2), the submittal of a schedule for two Regulatory Guide 1.97 variables (Standby Liquid Control Tank Level Instrumentation and Standby Liquid Control System Pump Flow) was deferred pending the development of plans and schedules to implement the ATWS (Anticipated Tranients Without Scram) rule, 10 CFR 50.62. This attachment describes the Authority's plans for these two variables.

1. Standby Liquid Control Tank Level Instrumentation

Our plans and schedules for complying with the requirements of 10 CFR 50.62 (the ATWS rule) were submitted to you in letter dated October 11, 1985 (Reference 6). As stated in that letter, the Authority is not altering the SLC system itself.

The equivalent SLC system flow requirement for FitzPatrick is 87.3 gallons per minute of a solution of thirteen weight-percent concentration of sodium pentaborate. This requirement will be met using an enriched boron solution. To achieve the required enrichment, enriched boric acid will be added to natural borax to produce double enriched sodium pentaborate solution. The final solution will be obtained by mixing the existing natural sodium pentaborate with a proportional amount of double enriched sodium pentaborate solution. The existing SLC system is capable of handling the enriched solution with no changes to key SLC system process parameters (flow rates, discharge pressure, required NPSH, etc.).

Therefore, no changes to the existing SLC tank level instrumentation are considered necessary at this time to comply with the requirements of 10 CFR 50.62.

2. Standby Liquid Control System Pump Flow Instrumentation

As stated in the Authority's November 30, 1984 Regulatory Guide 1.97 implementation report (Reference 1), there is currently no direct indication of Standby Liquid Control System Pump Flow in the FitzPatrick Control Room. However, the SLC system pump discharge header pressure is displayed in the Control Room. This indication, along with several other indications available to the operator in the Control Room, provide reasonable assurance of proper SLC system operation. Specifically, six indications can be used to confirm proper operation:

1. Loss of continuity to squib valve annuciator.
2. Loss of amber "Squib-Valve-Ready" lights.
3. Illumination of SLC "Pump-Running" light.
4. SLC pump discharge header pressure greater than reactor pressure.
5. Decreasing SLC tank level.
6. Reactivity decrease in reactor as measured by neutron flux monitoring.

The Authority considers that these indications are adequate to confirm proper SLC system operation; additional instrumentation would not significantly increase the ability to detect system maloperation. The BWR Owners Group has taken the same position in their July 1982 report (Reference 12).

Therefore, no changes to the existing SLC system instrumentation are necessary to comply with either 10 CFR 50.62 (the ATWS Rule) or Regulatory Guide 1.97 Revision 2. The instrumentation is acceptable as installed.

ATTACHMENT 3 TO JPN-85-91

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
DPR-59

Comments on Preliminary Technical Evaluation Report (TER) regarding Conformance to Regulatory Guide 1.97 Revision 2 and Corrections to June 28, 1985 Letter Regarding Implemetation Schedule

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1. The preliminary TER did not consider the new schedule information included with Reference 2. The revised schedules submitted with Reference 2 and the information in this letter (and attachments) should be incorporated in the final report.

2. The Authority has noted an error in the cover letter of Reference 2. Specifically, the third paragraph on the first page should read:

   > "One new variable (Item 18) that was omitted from the prior schedule has been added. Installation schedules for one variable (Item 23) has been shortened one fuel cycle. Four variables (Items 1, 10, 21 and 22) have extended installation schedules."

   The revised implementation schedule included with Reference 3 (Attachment 1) is correct - no changes are necessary.

References for Attachments to JPN-85-91

1. NYPA letter, J. C. Brons to D. B. Vassallo, dated November 30, 1984 (JPN-84-77) regarding response to Generic Letter 82-33.

2. NYPA letter, J. C. Brons to D. B. Vassallo, dated June 28, 1985 (JPN-85-53) regarding revised schedule for implementation of Regulatory Guide 1.97.

3. NRC letter, D. B. Vassallo to J. C. Brons, dated November 5, 1985 transmitted EG&G preliminary Technical Evaluation Report regarding the same subject.

4. NRC Generic Letter 82-33 dated December 17, 1982 "Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability," Section 6, "Regulatory Guide 1.97 - Application to Emergency Response Facilities."

5. EG&G Preliminary Informal Report, EGG-EA-7040, dated October 1985, "Conformance to Regulatory Guide 1.97, James A. FitzPatrick Nuclear Power Plant."

6. NYPA letter, J. C. Brons to D. B. Vassallo, dated October 11, 1985 (JPN-85-73) regarding implementation of ATWS rule modifications.

7. PASNY letter, J. P. Bayne to D. B. Vassallo, dated April 21, 1982 regarding post-TMI requirements (Generic Letter 82-05).

8. PASNY letter, J. P. Bayne to T. A. Ippolito, dated June 5, 1981 regarding post-accident shielding analysis. Includes shielding analysis results, source terms used, and areas considered.

9. NRC letter, R. W. Starostecki to C. A. McNeill, Jr., dated April 26, 1983 regarding Inspection Report 50-333/83-07. Inspection conducted March 28-31, 1983 of actions taken to comply with the requirements described in NUREG-0737 Item II.B.2, "Design Review of Plant Shielding."

10. NRC letter, G. H. Smith to J. P. Bayne, dated April 29 , 1982 regarding I&E Inspection 50-333/82-03. Opened NRC Inspection Item

50-333/82-03-016 on Area Radiation Monitors System upgrade (Appendix B, Item 11).

11. NRC letter, T. T. Martin to C. A. McNeill, Jr., dated November 16, 1983 regarding Inspection 50-333/83-22. Closed NRC inspection Item 50-333/82-03-016 regarding Area Radiation Monitoring System upgrade study.

12. BWR Owners' Group Report dated July 1982, "Position on NRC Regulatory Guide 1.97 Revision 2."

13. James A. FitzPatrick Nuclear Power Plant updated Final Safety Analysis Report (FSAR) Section 3.9, "Standby Liquid Control System."

14. NYPA letter, J. P. Bayne to D. B. Vassallo, dated May 20, 1983 (JPN-83-45). Attachment 1 to this letter identifies equipment requiring qualification in accordance with 10 CFR 50.49.

15. NRC letter, D. R. Muller to J. C. Brons, dated December 11, 1985 transmits order modifying license to confirm additional license commitments on the Safety Parameter Display System (SPDS), Regulatory Guide 1.97 and Technical Support Center (TSC).

16. "Emergency Preparedness for What? (A Review of the TMI-2 Incident)" by Andrew P. Muller, Safety and Environmental Protection Division, Brookhaven National Laboratory (BNL 29257).