POWER AUTHORITY OF THE STATE OF NEW YORK

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October 22, 1979
JPN-79-65

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

Attention: Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Response to NRC Requirements Based on
Studies of TMI

Reference: Letter Darrell G. Eisenhut (NRC) to all
Operating Nuclear Power Plants, Dated
September 13, 1979

Dear Sir:

Enclosed as Attachment 1 to this letter are the Authority's proposed implementation commitments and schedules in response to the requirements of the referenced letter.

The Authority recognizes the importance of efforts to apply knowledge gained from the Three Mile Island accident and proposes an expedited schedule, wherever possible meeting the implementation schedules of Attachment 6 to the referenced letter.

In certain cases, where substantial engineering effort is required, final installation schedules for equipment have not been proposed, but will be as soon as reliable estimates are available.

Very truly yours,

Paul J. Early
Assistant Chief Engineer-Projects

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ADD:
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ATTACHMENT I

THREE MILE ISLAND LESSONS LEARNED

COMMITMENTS*

*When scheduled dates are neither specifically provided herein nor reserved for later identification, the dates identified by NRC in the September 13, 1979 letter will be met.

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ATTACHMENT I

THREE MILE ISLAND LESSONS LEARNED COMMITMENTS

2.1.1 - Emergency Power Supply for Power Operated Relief Valves and Pressurizer Level Indicator.

As discussed in NEDO-24708, natural circulation in the BWR is strong and inherent in all off-normal modes of operation, independent of any powered system, as long as sufficient inventory is maintained. This is because even in normal operation the BWR is essentially an augmented natural circulation machine. Because the BWR operates with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. Thus, there is no need for emergency power to maintain natural circulation or to keep the system pressurized.

The power-operated relief valves in BWR's are already powered by emergency power. They have no blocking valves.

The reactor vessel level indication instrument channels for safety system activation and control are already powered by emergency power.

For the reasons stated above, the Authority believes no action is necessary in response to recommendation 2.1.1 for the FitzPatrick Plant.

2.1.2 - Relief and Safety Valves Tests

The BWR design basis includes no transients or accidents in which two-phase flow or subcooled liquid flow at high pressure is calculated or expected. In determining the need for special testing of BWR safety and relief valves it is essential to consider the service duty to which the primary system relief and safety valves of the BWR are exposed, and the consequences of maloperation of these valves. Relief valves are routinely used to mitigate the effects of system transients. A stuck-open valve is not an event of great significance in a BWR: in 300 reactor years of experience, 50 cases have occurred; in 3 such cases, the safety relief valves passed two-phase flow. Tables 1 and 2 summarize the experience to date. This experience, as will be explained, clearly shows that there is no need for an extensive testing program for BWR safety and relief valves.

A) BWR Safety and Relief Valves

Table 2.1-3 of NEDO-24708 shows the complement of safety and relief valves for all domestic operating BWRs. The FitzPatrick plant has eleven (11) safety/relief valves (S/RV) designed to mitigate the effect of system transients. Their discharges are piped to the containment suppression pool.

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This heat sink prevents significant containment heatup. Compliance of a system transient by a stuck-open valve has essentially no effect on reactor vessel water level measurement or on forced or natural circulation capability. The flow through the valve is saturated steam. If the valve cannot be closed by operator action the plant can be shut down using normal operating procedures.

B) Two-Phase Flow

Expected operating conditions and transients do not include two-phase flow through S/RV's. However, in 3 incidents, circumstances combined to cause high pressure water to flow down the steam lines and a steam/water mixture to flow through the valves. A summary of these events is given in Table 2. In these events, Electromatic relief valves and direct acting safety valves were actuated, discharged a steam/water mixture and reclosed, indicating that the flow did not cause a stuck-open valve condition. However, following these events, high water level trips were added to the FiztPatrick plant. Since all the S/RVs are piped to the suppression pool, direct pressurization of the drywell is minimal.

C) Valve Qualification

Three-stage Target Rock S/RVs were subjected to restricted flow steam tests to qualify the set-point and valve opening time delay. Solenoid valves (used during power actuation) are qualified by autoclave test for the LOCA environment. Satisfactory valve operation is indicated by field service.

D) Field Experience

Since 1971 there have been 50 events in BWR plant operation wherein S/RV's have stuck open (Table 1). In each of these cases the reactor was depressurized, the stuck valve was repaired or replaced, and the plant was placed back into service.

Although a stuck-open S/RV is ordinarily of no safety concern, programs are underway to reduce the frequency of such events. From Table 1 it is seen that the total number of S/RV blowdowns has steadily decreased since the mid-70's. The improvement in the number of S/RV blowdowns as a factor of number of S/RV's in service has been even more dramatic. From Table 2 it is seen that experience with 2-stage Target Rock relief valves has been good. At the FitzPatrick plant 9 out of the 11 S/RVs have been modified to the 2-stage type and the remaining 2 will be modified during the 1980 refueling outage.

E) Summary

(1) BWR S/RV's are routinely tested for the only expected mode of operation (saturated steam), both by in-place functional tests and by frequent usage in mitigating plant transients;

(2) There is no design-basis transient or accident which requires S/RV's to pass two-phase or liquid flow at high pressure.

(3) Inadvertent passage of two-phase flow is not likely where high pressure feedwater and injection system are tripped by high vessel water level.

(4) In the three events wherein BWR S/RV's did pass two-phase flow, the valves reclosed.

(5) The consequences of a stuck-open valve are minimal and reactor shutdown is uncomplicated, as proven by numerous filed occurrences. The procedures for responding to a stuck-open relief valve includes the opening of additional relief valves. This is no concern for core uncovery, and the valve need not pass two-phase flow. Improvement from 3-stage to 2-stage topworks on S/RVs will reduce the frequency of such events.

2.1.3.a - Direct Indication of Valve Position

The Authority's program to implement the captioned NRC position calls for engineering review to be completed by December 1, 1979 and final implementation during the Spring 1980 scheduled refueling outage, dependent upon equipment availability.

2.1.3.b - Instrumentation for Detection of Inadequate Core Cooling

Additional hardware to identify inadequate core cooling on BWRs is not determined to be necessary at this time. Procedures will identify the diverse methods of determining inadequate core cooling, using existing instrumentation. The results of analysis being performed, in response to 2.1.9 will be factored into procedures as required, after the analysis is complete.

Because the BWR operates with both liquid and steam in the reactor pressure vessel, saturation conditions are always maintained irrespective of system pressure. Thus there is no need for a subcooling meter in the BWR.

2.1.4 - Diverse Containment Isolation

A review of the FitzPatrick Plant containment isolation systems by our Architect-Engineer to confirm that the existing design meets the captioned NRC position, is scheduled for completion by January 1, 1980.

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2.1.5.a - Dedicated H_2 Control Penetrations

A review of the FitzPatrick Plant purge system by our Architect-Engineer to confirm that the existing design meets the captioned NRC position, is scheduled for completion by January 1, 1980.

2.1.5.b - Not applicable to the FitzPatrick plant.

2.1.5.c - Recombiner Procedures

It is the position of the Authority that no action is required for this item at this time as the Nuclear Regulatory Commission has so indicated in the September 13, 1979 letter.

2.1.6.a - System Integrity for High Radioactivity

The Authority plans to implement measures for leak reduction for systems that could carry radioactive fluid outside of the containment. The Authority will also institute a program to include periodic leak checks on these systems, modifications identified as a result of the leak reduction program will be examined and a schedule for implementation will be proposed.

2.1.6.b - Plant Shielding Review

The Authority's program to implement the captioned NRC position calls for engineering review to be completed by April 1, 1980. More guidance may be sought regarding development of shielding source terms and allowable compartment radiation levels. Any plant modifications indicated by the review will be examined by May 1, 1980, and a schedule for implementation proposed at that time.

2.1.7.a - Auto Initiation of Auxiliary Feed

This NRC position does not apply to the FitzPatrick Plant design.

2.1.7.b - Auxiliary Feed Flow Indication

This NRC position does not apply to the FitzPatrick Plant design.

2.1.8.a - Post Accident Sampling

The Authority's program to implement the captioned NRC position calls for engineering review to be completed by April 1, 1980. Any plant modifications indicated will be examined at that time, and a schedule for implementation proposed. Procedures will be developed for post accident sampling after engineering review and implementation of necessary modifications.

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2.1.8.b - High Range Radiation Monitors

Interim procedures will be developed to estimate noble gas and radioiodine concentration, should the existing instrumentation go off scale. Based on availability and current state-of-the-art, high range radiation monitors will be procured and installed in the containment. Additional procedures will be developed based on the new hardware changes.

2.1.8.c - Improved Inplant Iodine Instrumentation

No additional hardware is necessary for improved inplant iodine instrumentation, as the existing plant equipment is more than adequate. The Authority will review the existing plant procedures and modify them as necessary to meet the NUREG position.

2.1.9 - Transient and Accident Analysis

This item is already covered in the responses being provided to the Commission through the Bulletin and Orders Task Force. Specific requirements are being developed in a continuing series of meetings between the BWR Utility Owners Group and the NRC Bulletin and Task Force. The implementation of special procedures and retraining will be done on a schedule consistent with those established with the Bulletin and Orders Task Force.

Addition Addendum Items to NUREG 0578

Instrumentation to Monitor Containment Conditions During Course of Accident (Containment Pressure, Water Level, and Hydrogen Monitor)

The Authority's program calls for engineering review of the captioned NRC position to be completed by March 1, 1980. Any plant modifications deemed necessary will be evaluated at that time, and a schedule for implementation proposed.

- RCS Venting

BWRs like the FitzPatrick Plant are provided with a number of power operated safety grade relief valves which can be used to vent the reactor pressure vessel in addition to the air operated vent valves on the reactor vessel head. The piping arrangement is such that accumulation of gases above this point in the vessel will not affect natural circulation cooling of the reactor core.

The power operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves has been provided in the FitzPatrick Final Safety Analysis Report.

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The Authority's position is that the requirement of single failure criteria for prevention of inadvertent actuation of these valves is not applicable to BWR's. These valves serve an important function in mitigating the effects of transients and also provides ASME code overpressure protection. Therefore, the addition of a second "block" valve to the vent lines could result in a less safe design and in some cases a violation of the ASME code requirements. Also, inadvertent opening of relief valve in a BWR is a design basis event and is a controllable transient (this is discussed in our position of NUREG-0578, Item 2.1.2).

In addition to the power-operated relief valves, FitzPatrick Plant includes various other means of high-point venting. Among these are:

1) Normally closed reactor vessel head vent valves, operable from the control room which discharge to the drywell equipment drain sump;

2) Normally open reactor head vent line, which discharge to a main steam line;

3) Main steam-driven Reactor Core Isolation Cooling (RCIC) System turbines, operable from the control room, which exhaust to the suppression pool;

4) Main steam-driven High Pressure Coolant Injection (HPCI) system turbines, operable from the control room, which exhaust to the suppression pool;

Although the power-operated relief valves fully satisfy the intent of the requirement, these other means also provide protection against the accumulation of noncondensibles in the reactor pressure vessel.

Because the relief valves, HPCI, and RCIC will vent the reactor continuously, and because containment hydrogen calculations in normal safety analysis calculations assume continuous venting, no special analyses are required to demonstrate "that the direct venting of noncondensible gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment".

In view of what has been stated above the Authority believes that adequate reactor coolant system venting is provided by the existing plant design.

2.2.1.a - Shift Supervisor Responsibilities

The Authority plans to review and update if necessary the administrative and management procedures to emphasize the duties, responsibilities and authority of the Shift Supervisor as delineated in NUREG 0578.

2.2.1.b - Shift Technical Advisor 1210 185

The Authority plans to hire qualified technical personnel to work on each shift as Surveillance Test Engineer to meet the requirement concerning operating experience assessment. The Shift Supervisor will be trained and qualified as necessary to satisfy the accident assessment function. However, as it is impossible to hire and/or train these people by January 1, 1980, the Authority plans to utilize plant engineers to be on call at a short time notice to be available at the plant during emergency. The Authority plans to meet the requirement of this NUREG position by January 1, 1981. However, it is anticipated that fully training the Shift Supervisors will require continuous action until January 1, 1982.

2.2.1.c - Shift and Relief Turnover Procedures

The Authority plans to review and revise plant procedures as necessary to assure that adequate coverage exists during shift and relief turnover.

2.2.2.a - Control Room Access

The Authority plans to review plant procedures and revise them as necessary to assure that access to the control room is limited to those persons necessary for the safe command and control of operations.

2.2.2.b - Onsite Technical Support Center

A Onsite Technical Support Center exists for the FitzPatrick plant in the onsite emergency center. The permanent location for the Technical Support Center with the filtered ventilation system and necessary communication links and monitoring capability for the critical reactor parameters will be established within the restraint of construction and instrumentation availability.

2.2.2.c - Onsite Operation Support Center

The Authority plans to utilize the visitor's gallery and corridor to the control room as an operation support center.

Item Covered by Enclosures 7 and 8 to the September 13, 1979 NRC letter

Near Term Emergency Preparedness Improved Implementation

1) Upgrade Emergency Plan

The Authority has initiated action to upgrade the emergency plan to meet the requirements of RG. 1.101, Revision 1.

2) Short Term Actions Recommended by Lessons Learned Task Force

Items covered under this heading, namely 2.1.8.a, b, and c are already addressed and as such no action plan is indicated under this heading.

3) Emergency Operation Center for Federal, State and Local Officials

The temporary emergency operating center for federal, state and local officials exists for the FitzPatrick Plant. The

Authority plans to review and take necessary steps to meet the long term requirement.

4) Improved Off Site Monitoring Capability

It is the Authority's position that the FitzPatrick Plant is already in compliance as large numbers of TLD's have been distributed throughout the surrounding area of the plant site to monitor off site radiation exposure to the public.

5) Adequacy of State/Local Plans

The Authority has reviewed the adequacy of the state plan and has suggested action in upgrading the local plans. New York State has an NRC approved emergency plan for dealing with radiological emergencies in nuclear power plants.

6) Conduct of Test Exercises

It is the position of the Authority that the plant continue the present emergency plan testing as specified in the technical specifications and any augumentation that will be necessary will be implemented within the 5 year time schedule.

TABLE 1

S/RV BLOWDOWNS IN BWR OPERATION

YEAR	3-STAGE TARGET ROCK: TOTAL BLOWDOWNS	3-STAGE TARGET ROCK: STUCK OPEN FOLLOWING DEMAND	3-STAGE TARGET ROCK: # OF VALVES IN SERVICE	2 STAGE TARGET ROCK: TOTAL BLOWDOWNS	2 STAGE TARGET ROCK: # OF VALVES IN SERVICE	CROSBY-OKANO-DIKKERS: TOTAL BLOWDOWNS	CROSBY-OKANO-DIKKERS: # OF VALVES IN SERVICE	TOTAL S/RV BLOW-DOWNS	TOTAL S/RVs IN SERVICE	TOTAL BLOWDOWNS DIVIDED BY TOTAL VALVES IN SERVICE
1971	2	2	14					2	4	0.5
1972	1	1	23					1	23	0.04
1973	1	1	56					1	56	0.02
1974	10	1	108					10	108	0.09
1975	7	0	127					7	127	0.06
1976	11	1	149					11	149	0.07
1977	9	4	157					9	157	0.05
1978	5	3	157	0	11	0	35	5	203	0.02
1979 to Sept.	4	1	132	0	36	0	52	4	220	0.02

NOTE: The above table does not include Dresser Safety Valves (unpiped discharge) or "Electromatic" relief valves. See Table 2 for information on this equipment.

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TABLE 2

BWR EVENTS IN WHICH TWO-PHASE FLOW OR

LIQUID PASSED THROUGH SAFETY/RELIEF VALVES

DRESDEN 2 - JUNE 5, 1970

During the course of the initial test program on Dresden 2 with the unit operating at 75% power, a spurious signal in the reactor pressure control system occurred. This spurious signal resulted in sumultaneous opening of the control and the turbine bypass valves with resultant turbine trip, reactor scram, and main steamline isolation.

In response to the initial and expected water level drop, the operator switched to manual control of the feedwater system and began filling the reactor vessel at the maximum rate. Water level misinterpretation led to reactor water overflowing into the main steam lines. A pressure surge resulted in the main steam lines when relief valves were cycled. This momentarily opened one of the safety valves, resulting in a discharge directly to the containment (unpiped discharge). The fluid impinged upon the lifting levers of two other safety valves causing these safety valves to cock slightly open. The water-steam mixture from the two safety valves pressurized the primary containment. As a result, the containment was pressurized to an estimated 20 psig and an estimated temperature of approximately 300°F. Damage within the drywell was generally limited to over-heating of most of the flux monitoring instrumentation cables and water impingement on insulation. At no time during the event was there difficulty maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

DRESDEN 2 - DECEMBER 8, 1971

Unit 3 was operating about 98% power on December 8, 1971, when the plant was shut down due to a reactor low water level scram. The scram resulted from a condensate/condensate booster pump trip and the subsequent trip of two reactor feed pumps on low suction pressure. Following the scram, the standby feed pump started. The vessel was overfilled and the steam lines flooded. Due to a pressure surge in the main steam lines, a safety valve lifted causing discharge directly to the containment (unpiped discharge). Pressurization of the containment continued as high as 20 psig. Inspections showed that the high humidity and temperature in the drywell following the release to the containment damaged LPRM cables, which required replacement. Other results of the discharge from the safety valve included damage to an electromatic relief valve controller, damage to insulation near the safety valve, r ured paint on the drywell walls, and a damaged ventilation duct. There was never any concern for maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.

KRB (GERMANY) - JANUARY 13, 1977

The unit was operating at 100% power when a bus on two of its 200 KV lines opened. The plant was scrammed and isolated. Manual feedwater control was initiated which resulted in flooding of the steam lines. Safety valves opened and discharged water, steam and two-phase media. The valves discharged directly to the containment (unpiped discharge). The safety valves opened and reclosed several times. Because of the unique piping arrangement (which is not present in any US-BWR), reaction forces of the discharging valves caused or contributed to a pipe rupture in two of the fourteen flanged nozzles by which the valves are connected to a U-shaped header. At no time during the event was there concern for maintaining adequate water supply to the reactor core, and there was no question of adequate core cooling.