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P.O. BOX 529100 MIAMI, FL 33152

FLORIDA POWER & LIGHT COMPANY

REGION II ATLANTA

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April 24, 1979
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Mr. James P. O'Reilly, Director, Region II
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

Re: RII:JPO
50-250, 50-251
IE Bulletins 70-06, 79-06A

Florida Power & Light Company has reviewed IE Bulletins 79-06 and 79-06A, and a response, numbered to correspond to Bulletin 79-06A, is attached.

The response reflects the considerable effort ongoing at FPL due to the implications of the Three Mile Island incident. In order to follow up on the open commitments appearing in the response, a status report will be issued no later than May 31, 1979. In the meantime, we are available for further discussions with your staff, if you feel such discussions would be of benefit.

Very truly yours,

Robert E. Uhrig
Vice President
Advanced Systems & Technology

REU/MAS/ms

Attachment

cc: Robert Lowenstein, Esquire

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Response 1

Licensed operators, plant management, and supervisors with operational responsibilities have completed review/instructional sessions which encompassed the elements of Items 1.a and 1.b. These sessions were supplemented by a special on-site presentation by the USNRC. An outline of the material presented and the personnel in attendance is documented in plant records.

Response 2

Review of the actions required by operating procedures for coping with transients and accidents is conducted as part of the training required by Technical Specification 6.4.1. The specifics identified in Items 2.a, 2.b, and 2.c were included in the review/instructional sessions discussed in the response to Item 1. Ongoing review by both Florida Power & Light Company and our NSSS vendor may identify additional actions. If further actions are required, these actions will be expeditiously incorporated into plant operating procedures and training programs.

Response 3

Turkey Point Units 3 & 4 do use low pressurizer water level coincident with low pressurizer pressure for automatic initiation of safety injection. Upon notification by our NSSS vendor of a potential concern with this actuation logic, administrative changes were implemented to require that operators manually actuate the safety injection system if the reactor coolant pressure reaches the low pressure setpoint (exclusive of pressurizer level). In addition to directives placed in the control room, all operators attended briefing sessions during which the requirement to manually activate safety injection was thoroughly discussed. Presently, Florida Power & Light Company (FPL) is evaluating, with the NSSS supplier, control circuit logic modifications necessary to automatically actuate safety injection exclusive of pressurizer level. We anticipate making the necessary modifications upon completion of satisfactory reviews of the intended modifications by the on-site and off-site review groups (the reviews are required by Technical Specifications), and after obtaining any necessary changes to the plant Technical Specifications.

At this time, FPL views operation of the Turkey Point Units with the pressurizer level bistables tripped as undesirable. To operate in this manner increases the potential for the occurance of an undesirable transient (i.e. the greater risk associated with the highly increased probability for occurance of a full load rejection with SI actuation, resulting in immediate containment isolation, loss of charging, loss of main feedwater flow, transfer of auxiliaries to offsite power, etc., outweigh the potential risk associated with a stuck open pressurizer relief valve). Because of our unresolved concern for safety and the potential for equipment damage, we feel that operation with the pressurizer pressure bistables tripped is not prudent at the present time.

Because of the large amount of time available to initiate SI for an event of the TMI-2 type, manual actuation of the SI system, exclusive of pressurizer level, coupled with the system's normal automatic initiation provides conservative assurance that transients can be accommodated until modifications can be made to automate the actuation.

Operations personnel have been issued special instructions directing them to manually initiate safety injection in the event a transient is experienced where the pressurizer pressure drops to the safety injection set point (1715 psig) on 2 out of 3 channels whether or not the pressurizer level remains above the safety injection initiation set point (5%).

Response 4

The containment isolation system is designed to limit the leakage of radioactive materials through fluid lines penetrating the containment building. All fluid lines penetrating the containment, whose function does not degrade needed safety features or core cooling capability are isolated as follows:

a. Upon automatic initiation of safety injection, or manual Phase A isolation.

b. By at least one locked closed valve for those lines which are not in use during normal plant operation.

c. By at least one check valve for those line in closed systems with flow into containment.

d. Component cooling water supply and return to the reactor coolant pumps is isolated automatically upon 2/3 Hi coincident with 2/3 Hi-Hi containment pressure, or manually (Phase B isolation).

e. The containment ventilation system is isolated upon initiation of safety injection and/or Phase A or B isolation and also upon high particulate or gaseous radioactivity in containment.

f. The main steam isolation and bypass warmup valves are isolated manually or automatically upon 2/3 Hi-Hi coincident with 2/3 Hi containment pressure or Hi steam line flow coincident with low steam generator pressure or low T_{avg}.

Operations personnel have been issued special instructions advising them that if a transient occurs where the pressurizer pressure drops to the safety injection initiation set point (1715 psig) on 2 out of 3 channels and any of the following conditions are experienced, it may be necessary to initiate phase A Containment Isolation:

Conditions:

(1) Rising containment sump level.

(2) Increasing containment pressure.

(3) Increasing containment activity as indicated on either the Process Radiation Monitoring System or the Area Radiation Monitoring System.

Thus, our current design and procedures provide for containment isolation and include provisions to keep containment isolation from being degraded by reset of initiating signals.

Response 5

The auxiliary feedwater system is automatically initiated, however, the feedwater regulator valves are modulated by operator action from the control room to maintain steam generator level. Normal practice and established procedures dictate that an operator (as a primary and essential function) monitor and maintain steam generator level(s) during transients or accidents.

Response 6

Procedures currently exist which provide the information/actions listed in Items 6.a and 6.b. The specific procedures are included in the listing provided in Attachment 1. Additionally, these specifics were included in the review/instructional sessions discussed in the response to Item 1.

Response 7

a. Operations personnel have been instructed not to override automatic actions of Emergency Safeguards Features (ESFs) unless continued operation of ESFs would result in unsafe plant conditions. The necessity for this action was included in the review/instructional sessions discussed in the response to Item 1. Procedures will be revised accordingly.

b. Current operating procedures provide instructions which meet the intent of Bulletin item 7(b). However, termination of the HPSI pumps is based on plant conditions as opposed to a specified time of operation. Special operator briefings have been conducted in which the operator was instructed to carefully evaluate plant conditions before stopping any safeguards equipment. The operator briefings further instructed the operator to regain positive pressure control in the RCS before stopping HPSI pumps. The plant operators have been instructed on the importance of maintaining the RCS in a subcooled condition following a transient. Information relative to saturation temperatures and pressures have been made readily available to the operators for use in assessing transient plant conditions.

FPL has recently received additional clarifying instructions from the NSSS vendor regarding emergency procedures. FPL is reviewing these instructions as well as other information in connection with further detailed review of our emergency procedures. We will advise you of the results of this detailed review. In the interim we have concluded that our current procedures coupled with the special operator briefings and the attendant special instructions provide adequate assurance that the proper response will be made to plant transients.

c. Turkey Point emergency procedures call for stopping of all reactor coolant pumps once minimum conditions for their operation cannot be met.

We have recently received additional clarification from the NSSS vendor concerning their recommendations for operation of reactor coolant pumps during emergency transients, which appear to be in general agreement with our current operating procedures. However, we are reviewing the recommendations in conjunction with our detailed review of the Turkey Point emergency procedures and will advise the NRC of the final results of the review.

d. Review/instructional sessions as discussed in the response to Item 1 have been completed. Additionally, special instructions have been issued as described in the response to Item 3. A procedure review is in progress (reference Attachment 1) and those procedures needing changes will be revised accordingly.

Response 8

We have reviewed our administrative control of valves, locks and switches and believe that our current program is effective. We do however believe that potential areas for improvement will be noted during our procedure review which is in progress. (Reference Attachment 1). Any appropriate remedial action will be initiated upon completion of the review.

Response 9

a. The design of the containment isolation system is such that it is not reset by the elimination of the initiating signals, e.g., by resetting safety injection or by eliminating the isolation initiating signal. Containment isolation can only be reset by manually resetting lockout type relays in the containment isolation racks.

Control features are provided for the containment isolation valves such that:

(1) All valves with exception of the containment purge, instrument air bleed, main steam and containment sump discharge will remain in the closed position if the respective containment isolation is reset and the initiating trip signal no longer exists. The containment sump discharge isolation valves will return to their normal position by resetting the Phase A Containment Isolation, provided the initiating trip signal no longer exists. The containment purge and instrument air bleed

valves will return to their normal position by resetting the Containment Ventilation Isolation, provided the initiating trip signal no longer exists. The main steam isolation valves will reset (but can not physically return) to their normal position in the absence of the initiating trip signal. FP&L is proceeding to revise this control scheme such that these valves will remain closed upon resetting of the isolation signal and/or absence of the initiating trip signal. In the interim special instructions have been provided.

(2) The containment isolation signals override all other automatic control signals.

(3) Each valve can be opened or closed manually only if the containment isolation signals have been manually reset.

(4) Isolation of the containment ventilation system is initiated upon high particulate or gas radioactivity in containment as well as manual or automatic initiation of safety injection or manual initiation of Phase A of Phase B Containment Isolation.

b. The criteria for isolating lines which penetrate the containment is as described in the response to item 4.

c. Operability of the above features is assured by periodic testing in accordance with technical specification requirements.

Response 10

A review of the procedures listed in Attachment 1 is in progress. Our review has addressed the concerns of Items 10.a, 10.b, and 10.c. We feel that the existing maintenance, test, and clearance procedures effectively address the Staff's concerns.

Response 11

As FPL understands this item, it addresses situations associated with significant release of radioactive material. Such release of radioactive material would be expected to be preceded by damage to fuel assemblies in the reactor. The plant currently has the means to identify such conditions.

When a condition such as described above is identified, the Turkey Point Plant Emergency Plan is put into effect and the Emergency Control Officer or his designated alternate is notified by onsite personnel. These officers are always available by telephone or beeper.

The Emergency Control Officer, who is located offsite, would notify NRC-I&E and give a report of the situation. Practice drills indicate that such notification can probably be made within one hour. We have adopted this system in order to allow onsite operators to devote maximum effort toward bringing the plant to a stable condition. Plant personnel periodically update the Emergency Control Officer on the status of the plant. The Emergency Control Officer would then periodically update NRC-I&E.

Considering the TMI-2 incident and the communication problems encountered by the NRC, FPL recognizes the need for the NRC to be fully and accurately informed about conditions at nuclear plants which may adversely affect the public health and safety. We believe that our established notification procedure meets the NRC concern for prompt notification.

We will continue to assess our ability to establish an "open continuous communication channel" which establishes direct voice contact with a responsible representative of the NRC as suggested in Item 11, and will inform you of our conclusions.

Response 12

The engineered safeguards are designed and analyzed to meet the limits of 10 CFR 50.46 which require that the hydrogen generation from clad water reaction in a LOCA be limited to less than 1% of the clad metal, and nowhere exceed 17% of the clad thickness.

These modes for removing hydrogen from the reactor coolant system are:

a. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if the reactor coolant pump is operating.

b. Hydrogen in the pressurizer vapor space can be vented by power operated relief valves to the pressurizer relief tank or by the pressurizer steam space sample line to the volume control tank.

c. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system. The waste gas system has six tanks with a capacity of 4400 SCF each.

d. In the event of a LOCA, hydrogen would vent with the steam to the containment.

If for some reason a non-condensible gas bubble becomes situated somewhere in the primary coolant systems, there are many options for continued core cooling and removing the bubble.

With a gas bubble located in the upper head several methods of core cooling are unaffected. The steam generator can be used to remove decay heat using reactor coolant pump forced flow or natural circulation. The safety injection system can be used to cool the core while venting through the pressurizer power operated relief valve. Core cooling by any of these methods can proceed indefinitely if the primary coolant pressure is held constant. If a lower system pressure is desired, a controlled depressurization will allow the bubble to grow slowly until it uncovers the top of the hot legs.

This controlled depressurization can be performed in two ways:

(1) If the reactor coolant pumps can be restored depressurization can be performed with a steam bubble in the pressurizer. Depressurization would be through the pressurizer power operated relief valve. Extra control is achieved with the pressurizer heaters and sprays if available. As the bubble grows to the top of the hot leg, small bubbles are carried through the system. Degassing is done with the spray line and/or the Chemical and Volume Control System. The steam generators will carry away decay heat.

(2) If the reactor coolant pumps cannot be operated or their operation is undesirable, the pressurizer can be made water solid with the safety injection pumps running and the power operated relief valve and/or vent valve open. Depressurization is controlled by judicious use of the various valves, lines and pumps available in the safety injection system and by adjusting the pressurizer relief valve and/or vent valve. As the bubble grows to the top of the hot leg, it slides across the hot leg and up into the steam generators. As depressurization continues the gas bubbles grow in the steam generators and upper head but the core remains covered and cooled by safety injection water. If there is enough gas, the pressurizer surge line would eventually be "uncovered". Some of the gas would burp into the pressurizer and out the valve. This burping process would continue until the system were at the desired pressure. At that time the current cooling mode could be continued or the system could be placed in an RHR mode (special care is needed for operation).

Note that a gas bubble cannot be located in the steam generator with the reactor coolant pumps running. If a gas bubble forms in the steam generator during natural circulation, the reactor coolant pumps could be turned back on for degassing or safety injection flow could be initiated with the power operated relief valve open.

Also note that the gas bubbles cannot uncover the core in the above depressurization schemes because they will always tend to float to the top of the system and cannot compress water.

A post-accident containment vent system is provided to facilitate controlled venting of the containment through HEPA and charcoal filters to the waste gas decay tanks and to the atmosphere. The post-accident containment vent system consists of a supply line through which air can be admitted to the containment, two containment dome collection headers feeding separate exhaust lines, and a HEPA and charcoal filter train which is connected to the waste disposal system vent header.

If the containment hydrogen concentration reaches 3.0 volume percent, pressurization of the containment via the service air supply line is started. When the containment pressure reaches 1.5 psig one waste gas compressor is started and its effluent directed to a gas decay tank. The venting process is stopped when the hydrogen concentration is reduced to 2.7 v/o.

ATTACHMENT 1: PROCEDURES IDENTIFIED FOR REVIEW

PROCEDURE NUMBER		TITLE
AP	0103.4	In Plant Equipment Clearance Orders
AP	0103.5	Administrative Control of Valves, Locks and Switches
AP	0103.6	Reportable Occurrences
AP	0190.19	Control of Maintenance on Nuclear Safety Related Systems
OP	0202.1	Reactor Startup, Cold Conditions to Hot Shutdown Conditions
OP	0202.2	Unit Startup, Hot Shutdown to Power Operation
OP	0205.2	Reactor Shutdown, Hot Shutdown to Cold Shutdown
OP	0208.1	Shutdown Resulting from Reactor Trip or Turbine Trip
OP	0208.3	Annunciator List - Panel A - Reactor Coolant
OP	0208.4	Annunciator List - Panel B - Reactor
OP	0208.5	Annunciator List - Panel C - Steam Generator and Reactor Trips
OP	0208.6	Annunciator List - Panel D - Condensate and Feedwater
OP	0208.7	Annunciator List - Panel E - Turbine Generator
OP	0208.8	Annunciator List - Panel F - Electrical
OP	0208.9	Annunciator List - Panel G - Miscellaneous
OP	0208.10	Annunciator List - Panel H - Safety Injection and Auxiliary
OP	0208.11	Annunciator List - Panel I - Station Services
OP	0208.12	Annunciator List - Panel X - Common
OP	0208.13	Annunciator List - Waste/Boron Panels
OP	0209.1	Valve Exercising Procedure
OP	0209.2	Inservice Pump Testing Program Implementation Procedure for RHR, SIS, and CS Pumps
OP	0209.3	Inservice Pump Testing Program Implementation Procedure for Auxiliary Feedwater Pumps
OP	0209.4	Inservice Testing - Valve Seat Leakage Testing
MP	0729	Safety Related MOV Motor Maintenance
OP	1004.2	Reactor Protection System - Periodic Test
OP	1008.2	Excessive Reactor Coolant System Leakage

ATTACHMENT 1

PROCEDURE NUMBER	TITLE
OP 1008.3	Loss of Reactor Coolant Flow
OP 1008.4	Excessive RCS Activity
OP 1100.1	Reactor Coolant Pump Operation
OP 1108.1	Reactor Coolant Pump Off-Normal Conditions
OP 1200.1	Pressurizer Steam Space Venting
OP 1207.1	Pressurizer Safety Valve, Repair and Setting
OP 1208.1	Pressurizer Malfunction of Power Operated Relief or Safety Va
OP 1208.2	Pressurizer — Malfunction of Level Control
OP 1300.1	Pressurizer Relief Tank Operation
OP 1508.2	Steam Generator Tube Failure
OP 3104.1	Component Cooling Water System - Periodic Test of Pumps
OP 3108.1	Component Cooling System - Loss of Component Cooling Flow
OP 3208.1	Malfunction of Residual Heat Removal System
OP 3404.2	Intake Cooling Water System - Periodic Test of Pumps
OP 3408.1	Intake Cooling Water - Malfunction
OP 4004.1	Containment Spray Pumps - Periodic Test
OP 4104.1	Safety Injection System - Periodic Test
OP 4504.1	Accumulator Check Valves Backleakage - Periodic Test
OP 4704.1	Emergency Contain-ent Filter - System Operating Test and Insp
OP 4704.6	Emergency Containment Coolers - Periodic Test
OP 5110.1	WDS - Reactor Coolant Drain Tank Operation
OP 7304.1	Auxiliary Feedwater System - Periodic Test
OP 10107.1	Repair of Containment Purge Valves
OP 13108.1	Loss of Containment Integrity
EP 20003	Loss of Reactor Coolant
EP 20005	Main Steam Line Break or Feedwater Line Break
EP 20006	Loss of Feedwater Flow or Steam Generator Level