

REGULATORY DOCKET FILE COPY.

*Docket File
D-250*

JULY 7 1980

Docket Nos. 50-250
50-251

Dr. Robert E. Uhrig, Vice Pres.
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

SUBJECT: NRC STAFF EVALUATION OF FLORIDA POWER AND LIGHT COMPANY RESPONSES TO
IE BULLETINS 79-06A AND 79-06A, REVISION 1, TURKEY POINT PLANT, UNIT
NOS. 3 AND 4

We have reviewed the information provided by your letters dated April 24,
May 4, June 4, 18 and 25, 1979 in response to IE Bulletins 79-06A and 79-06A,
Revision 1 for the Turkey Point Plant, Unit Nos. 3 and 4. The enclosure
provides our evaluation of your responses with respect to their specificity,
completeness, and responsiveness to the bulletins. In this regard, we have
found that you have taken appropriate actions to meet the requirements of IE
Bulletins 79-06A and 79-06A, Revision 1.

It should be noted that the staff review of the Three Mile Island, Unit 2
accident is continuing. Consequently, other corrective actions may be
required at a later date. For example, IE Bulletin 79-06C was issued on
July 26, 1979, requiring new considerations for operation of the reactor
coolant pumps following an accident. Our reviews of the Westinghouse Owners'
Group response to Items 2 and 3 of Bulletin 79-06C (Westinghouse reports
WCAP 9584 and WCAP9600, respectively) are documented in NUREG-0623 and
NUREG-0611, respectively. You will be kept informed regarding the require-
ments for the Turkey Point Plant resulting from these reviews by separate
correspondence.

Sincerely,
Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Evaluation of Licensee's Responses
to IE Bulletins 79-06A and 79-06A
(Revision 1)

TAC NOS 1312415

TACS 1312415
ORB-1
Mr. Grotenhuis
5/23/80
6/23/80

SB PDD
P. O'Reilly
5/24/80

~~SB
W. Kane
5/13/80~~

with comments
IE
E. Jordan
5/13/80

ORB-1
S. Varga
5/13/80

8008130296

#5

Q

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

July 7, 1980

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50-251

Dr. Robert E. Uhrig, Vice Pres.
Advanced Systems and Technology
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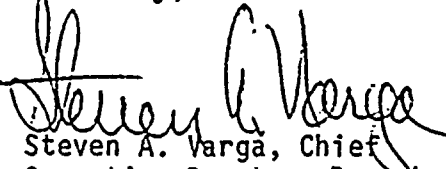
Dear Dr. Uhrig:

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IE BULLETINS 79-06A AND 79-06A, REVISION 1, TURKEY POINT PLANT, UNIT
NOS. 3 AND 4

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Sincerely,


Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
Evaluation of Licensee's Responses
to IE Bulletins 79-06A and 79-06A
(Revision 1)

July 7, 1980

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EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETINS
79-06A AND 79-06A (REVISION 1)

TURKEY POINT PLANT, UNIT NOS. 3 AND 4
DOCKET NOS. 50-250, 50-251

INTRODUCTION

By letters dated April 14, and April 18, 1979, we transmitted our Office of Inspection and Enforcement (IE) Bulletins No. 79-06A and 79-06A (Region 1), respectively, to Florida Power and Light Company (the licensee). These bulletins specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred on March 28, 1979 at Three Mile Island, Unit No. 2 (TMI-2). By letter dated April 24, 1979, the licensee provided its response to the aforementioned bulletins for the Turkey Point Plant, Unit Nos. 3 and 4. The licensee supplemented its response by letters dated May 4, June 4, 18 and 25, 1979, providing clarification and elaboration of certain of the Bulletin Items in response to our expressed concerns. Our evaluation of the licensee's response, as supplemented, is provided below.

EVALUATION

In this evaluation, the numbers correspond to the bulletin items and to the licensee's response to each Bulletin item.

1. In Bulletin Item No. 1, licensees were requested to review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 (issued to all licensees with Babcock & Wilcox (B&W)-designed plants for action, and to all other licensees for information) and the preliminary chronology of the TMI-2 accident included in Enclosure 1 to IE Bulletin 79-05A (same distribution as IE Bulletin 79-05).

- (a) This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident; (2) the apparent operational errors which led to the eventual core damage; (3) that the potential exists, under certain accident or transient conditions, to have a water level in the pressurizer simultaneously with the reactor vessel not full of water; and (4) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.
- (b) Operational personnel should be instructed to: (1) not override automatic action of engineered safety features unless continued operation of engineered safety features will result in unsafe plant conditions (see Section 7a.); and (2) not make operational decisions based solely on a single plant parameter indication when one or more confirmatory indications are available.
- (c) All licensed operators and plant management and supervisors with operational responsibilities were to participate in this review and such participation was to be documented in plant records.

An NRC briefing team provided a detailed review of the circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 accident included in Enclosure 1 of IE Bulletin 79-05A to a majority of the licensed operators and plant management. The briefing team consisted of an IE Project Inspector, an Operator Licensing Branch (OLB/NRR) representative, and the facility Principal/Resident Inspector. The NRC briefing also provided a detailed review of Items 1.a and 1.b of IE Bulletin 79-06A.

We consider the licensee review and participation in the NRC briefing to be an acceptable response to Bulletin Item No. 1.

- 2. Bulletin Item 2 requested licensees to review actions required by operating procedures for coping with transients and accidents, with particular

attention to (a) recognition of the possibility for forming voids large enough to compromise core cooling capability, (b) action required to prevent the formation of such voids, and (c) action required to enhance core cooling in the event such voids are formed. Emphasis in (a) was placed on natural circulation capability.

The licensee has reviewed the operating procedures for coping with transients and accidents and has found that the actions required by these procedures are sound; however, specific mention of the possibility for core voiding was lacking for the most part. Therefore, the licensee has issued a Special Instruction to the operators specifying the instrumentation to use in aiding the detection of voiding in determining whether or not natural circulation exists if reactor coolant pumps are inoperable, in taking corrective action to prevent void formation, and in enhancing core cooling under the natural circulation model of operation. A saturation curve for each unit has been placed in the Control Room for additional aid to help the operator maintain the reactor coolant in a subcooled state.

The Special Instruction tells the operator that natural circulation cooling can be enhanced when the primary water inventory is maintained and the reactor coolant is subcooled. To accomplish this, the pressurizer level should be at the normal no load level and not decreasing and the pressurizer pressures should be at ≥ 2000 psig. This pressure will result in at least 15°F subcooling at the core outlet at the maximum anticipated T_{HOT} of 620°F for natural circulation following a trip from 100% power.

Under these conditions, voids due to steam or noncondensable gas formation will not be present, so the operator knows that, with indicated level in the pressurizer, the reactor coolant system is in a non-voided condition.

The operator has been instructed that, for natural circulation to occur, a heat sink must be present and that this heat sink is maintained by maintaining the level in the steam generators at a point above the top of the tubes - that is, in the narrow range. After assuring that the above

conditions exist, the operator has been instructed to verify that natural circulation exists by determining the RCS ΔT , which should be less than the full load ΔT of 56°F. This is done by using the wide range temperature indication and subtracting T_{cold} from T_{hot} . A constant T_{hot} and T_{cold} also tells the operator that heat is being removed. Another indication that natural circulation is being maintained is steam pressure remaining constant or decreasing at the same rate as primary temperatures while maintaining steam generator level with continuous auxiliary feedwater flow. If natural circulation stopped, steam pressure would fall rapidly as the steam generator cooled, and steam generator level would rise, assuming continuous auxiliary feedwater flow.

If natural circulation is not indicated, measures to take to restore circulation flow include repressurization above T_{hot} saturation pressure by the use of pressurizer heaters and reestablishing steam generator level to the narrow range level in at least one steam generator.

The operators have also been instructed in the use of T_{hot} and in-core thermocouples in determining maximum saturation pressure for use in depressurizing to the degree necessary to achieve safety injection flow to the core, if necessary to restore pressurizer level. This instruction applies to the situation where a leak has occurred which, combined with coolant pump seal leakage, exceeds charging pump capacity.

Operating procedures have been revised to reflect the special instructions already given to the operators.

In addition, the licensee participated, as a member of the Westinghouse Owners Group, in the effort to develop generic guidelines for emergency procedures. In our November 5 and December 6, 1979 letters to the Owners Group, we approved the Westinghouse generic guidelines regarding small break LOCAs for implementation by licensees with Westinghouse-designed reactors. The Owners Group, in conjunction with Westinghouse, has also developed generic guidelines for emergency procedures regarding natural circulation. These generic guidelines were submitted on December 28, 1979, as part of the Owners Group response to the requirements of Item 2.1.9 of NUREG-0578 regarding inadequate core cooling. In order to

two-out-of-three low pressurizer pressure condition regardless of the pressurizer level.

We consider the licensee's response to Bulletin Item No. 3 acceptable.

4. Bulletin Item No. 4 requested that licensees review the containment isolation initiation design and procedures, and implement all changes necessary to permit containment isolation, whether manual or automatic, of all lines whose isolation would not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

Initiation of safety injection at Turkey Point by automatic or manual actuation signal actuates Phase A isolation of containment. Phase A isolates all non-essential process lines, but does not affect safety injection, containment spray, auxiliary cooling, or steam and feedwater systems. Therefore, Phase A isolation does not degrade needed safety features or cooling capability including the operation of reactor coolant pumps. Phase B isolation of containment is actuated by high-high containment pressure. Phase B isolation isolates all remaining process lines except safety injection, containment spray, and auxiliary feedwater. Although operation of the reactor coolant pumps cannot continue for very long when Phase B isolation stops component cooling water to the pump seals and motor bearings, the high containment pressure or need for containment spray would indicate a large rapid blowdown of the primary system. In that event, the reactor coolant pumps would not be of any use until after longer term reflooding had taken place.

We find that the licensee's response has adequately addressed the concerns expressed in Bulletin Item No. 4.

5. In Bulletin Item No. 5, licensees with facilities at which the auxiliary feedwater system is not automatically initiated were requested to prepare and implement immediately procedures which required the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generator(s) for those

transients or accidents, the consequences of which could be limited by such action.

The auxiliary feedwater system is automatically initiated at Turkey Point but operator action is required in order to ensure adequate flow. The operator must open the feedwater regulating valves (normally maintained closed) to obtain auxiliary feedwater flow to the steam generators. We consider this acceptable since all the applicable controls are located in the control room.

We find that the licensee's response has adequately addressed the concern expressed in Bulletin Item No. 5.

6. Bulletin Item No. 6 requested that licensees prepare and implement immediately procedures which:

- (a) Identified those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators could utilize to determine that the pressurizer power-operated relief valve(s) are open, and
- (b) Directed the plant operators to manually close the power-operated relief block valve(s) if the reactor coolant system pressure had been reduced to below the set point for normal automatic closure of the power-operated relief valve(s) and the valve(s) remained stuck in the open position.

The licensee reviewed the applicable Turkey Point procedures and determined that no changes or revisions were needed to comply with Bulletin Item Nos. 6.a and 6.b.

Based on our review, we find that the licensee's response to Bulletin Item No. 6 is acceptable.

7. In Bulletin Item No. 7, licensees were requested to review the action directed by the operating procedures and training instructions to ensure that:

(a) Operators do not override automatic actions of engineered safety features, unless continued operation of engineered safety features would result in unsafe plant conditions. For example, if continued operation of engineered safety features would threaten reactor vessel integrity, then the high pressure injection (HPI) system should be secured (as noted in b(2) below).

(b) Operating procedures currently, or are revised to, specify that, if the (HPI) system had been automatically actuated because of a low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing for 20 minutes or longer at a rate which would assure stable plant behavior, or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees Fahrenheit below the saturation temperature for the existing RCS pressure. If 50 degrees subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated. The degree of subcooling beyond 50 degrees and the length of time HPI has been in operation shall be limited by the pressure/temperature considerations for the vessel integrity.

(c) Operating procedures currently, or are revised to, specify that, in the event of HPI initiation with reactor coolant pumps (RCPs) operating, at least one RCP shall remain operating for two-loop plants and at least two RCPs shall remain operating for 3 or 4 loop plants, as long as the pump(s) is providing forced flow.

(d) Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also

examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

The licensee has agreed to utilize the bulletin criteria for override of emergency safeguard features. This constitutes an acceptable response to Bulletin Item No. 7.a.

In response to Bulletin Item No. 7.b, the licensee participated in the effort by the Westinghouse Owners Group, in conjunction with Westinghouse, to develop generic guidelines for emergency procedures. In our November 5 and December 6, 1979 letters to the Owners Group, we approved generic guidelines for emergency procedures regarding small break LOCAs for implementation by licensees with Westinghouse-designed operating plants. These approved guidelines include the following criteria (taken from the enclosure to our letter of December 27, 1979) for termination of safety injection:

- (1) The reactor coolant system pressure is greater than 2000 pounds per square inch gauge and increasing, and
- (2) The pressurizer water level is greater than the programmed no-load water level, and
- (3) The reactor coolant indicated subcooling is greater than (insert plant-specific value, which is the sum of the errors for the temperature measurement system used and the pressure measurement system translated into temperature using the saturation tables), and
- (4) The water level in at least one steam generator is stable and increasing, as verified by auxiliary feedwater flow to that unit. Auxiliary feedwater flow to the unaffected steam generator should be greater than (a value in gallons per minute sufficient to remove decay heat after 20 minutes following reactor trip) until the indicated level is returned to within the narrow range level instrument.

satisfy NUREG-0578 requirements, the licensee has incorporated the guidelines into the Turkey Point procedures (small break LOCA guidelines by January 1, 1980 and inadequate core cooling guidelines by January 31, 1980). The Office of Inspection and Enforcement will verify that acceptable guidelines have been properly implemented. Procedures based on these generic guidelines represent an acceptable method of complying with Bulletin Item No. 2.

We find that the licensee has provided an acceptable response to Bulletin Action Item No. 2.

3. Bulletin Item No. 3 requested that licensees with facilities that used pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system trip the low pressurizer level setpoint bistables such that, when the pressurizer pressure reached the low setpoint, safety injection would be initiated regardless of the pressurizer level. The pressurizer level bistables could be returned to their normal operating positions during the pressurizer pressure channel functional surveillance tests.

On April 24, 1979, the licensee notified us that Turkey Point Units 3 and 4 do use lower pressurizer water level coincident with low pressurizer pressure for automatic initiation of safety injection. Upon notification by the 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. On May 4, 1979, we issued Amendment No. 48 and 40 to the Turkey Point operating licenses. These licensee amendments approved the design change to the safety injection initiation logic which the licensee had proposed. This design change consisted of modifying the safety injection initiation system logic so that safety injection will be initiated on a

Details of our evaluation of this issue are included in the report (NUREG-0611) of our generic review of Westinghouse-designed operating plants.

Pending verification by the Office of Inspection and Enforcement that the approved Westinghouse generic safety injection termination criteria have been properly incorporated in the Turkey Point plant procedures, we find that the licensee's actions with regard to Bulletin Action Item No. 7.b are acceptable.

Another issue on which the Westinghouse Owners Group worked, in conjunction with Westinghouse, to achieve resolution with the staff was the matter of reactor coolant pump operation following a small break LOCA (Bulletin Item No. 7.c). On July 26, 1979, IE Bulletin 79-06C superseded Item No. 7.c of Bulletin 79-06A. Bulletin 79-06C required that, as a short-term action, licensees were to trip all reactor coolant pumps after an initiation of safety injection caused by low reactor coolant system pressure. In its August 31, 1979 response to Bulletin 79-06C, the licensee stated its conformance with this requirement. This action was to remain in effect until the results of analyses specified in Bulletin 79-06C had been used to develop new guidelines for operator action.

We have completed our review of the reactor coolant pump trip issue with the Owners Group. The generic guidelines for emergency procedures regarding small break LOCAs, which we approved in our November 5 and December 6, 1979 letters to the Owners Group, contain the approved pump trip criteria for Westinghouse-designed operating plants. Basically, they are as follows:

- (1) Stop all reactor coolant pumps after high pressure safety injection pump operation has been verified, and when the wide range reactor pressure is at (plant-specific pressure derived from secondary system relief capacity, primary-to-secondary system pressure difference, and instrument inaccuracies).

Appropriate cautions have been included in the guidelines regarding isolation of component cooling water to the reactor coolant pumps and maintaining seal injection flow to preclude pump damage due to inadequate cooling. The details of our review of the pump trip issue are reported in NUREG-0623.

Pending confirmation by our Office of Inspection and Enforcement that the licensee has incorporated the pump trip criteria as specified in the approved Westinghouse generic guidelines into the Turkey Point plant procedures, we find the licensee's response to Bulletin Item No. 7.c acceptable.

In response to Bulletin Item No. 7.d, the licensee issued special instructions to Turkey Point operations personnel which cautioned against overreliance on pressurizer level indication, and recommended examination of other plant parameters in assessing water inventory and plant conditions. In addition, the concern expressed in this bulletin item was incorporated in the licensee's operator training program. In its June 4, 1979 letter, the licensee supplemented its original response to identify the specific plant parameters to be used in assessing water inventory and plant conditions. The licensee also stated that the applicable procedures were revised to reflect the above-mentioned considerations. We find these actions to be an acceptable response to Bulletin Item No. 7.d.

8. Bulletin Item No. 8 required that licensees review alignment requirements and controls for all safety-related valves necessary for proper operation of engineered safety features. In their June 4, 1979 response, the licensee stated that they have reviewed the administrative control of valves, locks and switches and believe that the current program is effective. All valves on the "Valve Lock and Switch List" have been field verified. Additionally the NRC Inspector randomly selected safety systems and field verified proper system line-up during his inspection from 5/1/79 - 5/3/79. All procedure revisions have been completed.

Based on our review, we find the licensee's response to Bulletin Item No. 8 acceptable.

9. In Bulletin Item No. 9, licensees were requested to review their procedures to assure that radioactivity will not be inadvertently released from containment. Particular emphasis was placed on the resetting of engineered safety features (ESFs) and the effects of this action on valves controlling the release of radioactivity.

In the letter dated June 25, 1979 the licensee response was revised to include the following:

Prior to the start-up of Turkey Point Unit 4, a special interim instruction will be implemented to require:

- (1) that controllers for the sump pump discharge valves will be placed in the shut position prior to resetting phase A containment isolation,
- (2) that the controllers for the containment purge and instrument airbleed isolation valves will be placed in the shut position prior to resetting containment ventilation isolation, and
- (3) that the main steam isolation valves' control switches will be placed in the closed position after any valid MSIV isolation signal.

NOTE: Response 9 (1) has been revised and response 9 (3) has been added because MSIV closure is not related to phase A containment isolation. Placing the control switches in the closed position will prevent these valves from re-opening, even in the absence of the steam line isolation signal, and the absence of a pressure differential across the valves.

We find that the licensee has adequately addressed the concerns expressed in Bulletin Item No. 9.

The staff's implementation of Item 2.1.4 of NUREG-0578 provides further assurance that the inadvertent release of radioactivity from containment upon resetting of emergency safeguard features will be precluded. Our

review of NUREG-0578 Item 2.1.4 implementation will be reported in a separate document.

10. Action Item No. 10 of Bulletin 79-06A required that licensees review and modify, as necessary, maintenance and test procedures for safety-related systems to ensure that they require that: (a) redundant systems are operable before a system is taken out of service, (b) systems are operable when returned to service, and (c) operators are made aware of the status of these systems.

The licensee has completed the procedure review (reference letter L-79-100, dated April 24, 1979) and has identified only a few areas where the procedures may be enhanced. The staff concerns in Items a, b, and c are addressed generically in AP 0103.4, "In-Plant Equipment Clearance Orders" and AP 0190.19, "Control of Maintenance on Nuclear Safety Related Systems."

The changes identified in the review to enhance the procedures have been processed. All of these revisions are completed.

- a. Plant procedures have been reviewed to ensure that they require verification of the operability of redundant safety related systems prior to the removal of any safety related system from service.

All clearances requested on nuclear safety related systems, or equipment listed in Technical Specification 3.0, Limiting Conditions for Operations, are given a review before issuing. This review is conducted by the Nuclear Plant Supervisor or the Nuclear Watch Engineer and a member of the plant operating staff, holding an active Senior Reactor Operator's License. This review ensures that the Minimum Equipment List for the Limiting Conditions for Operation remains satisfied.

The Minimum Equipment List is reviewed each shift and completed to indicate those components/systems which are operable and available. The status of the component/ system, i.e., operability, is based on

the interpretation by the USNRC that performance of a surveillance requirement within the specified time interval shall constitute compliance with operability requirements for a Limiting Condition for Operation and associated action statements unless otherwise required by the specification. Surveillance Requirements do not have to be performed on inoperable equipment.

- b. Plant procedures have been reviewed to ensure that they require verification of the operability prior to being declared ready-for-service/operable. The procedures require:
 - 1. All clearances to be properly released.
 - 2. All tags to be removed and valves, switches, etc. to be in their normal lineup.
 - 3. The system has been tested to the extent that it is evaluated safe to return to service, specifically;
 - (1) for pumps and valves which fall under the Inservice Inspection Program, that appropriate retests are performed satisfactorily, and
 - (2) for equipment which falls under the Technical Specification Limiting Conditions for Operation, that appropriate surveillance tests are performed satisfactorily.
- c. The removal of safety-related equipment from service for maintenance or test is required by procedure to be reviewed and approved by the on duty nuclear watch supervisor or nuclear watch engineer, in addition to a licensed senior reactor operator. A log of equipment clearance orders is required by procedure to be maintained in the control room. In addition any safety-related component that is removed from service must be logged in the equipment out of service log book which is also maintained in the control room.

Licensee procedures require that the equipment clearance order log book be reviewed periodically by the nuclear watch supervisor and the nuclear watch engineer; and that the equipment out of service log book be reviewed by the oncoming nuclear watch supervisor and the nuclear watch engineer each shift.

Based on our review, we find that the licensee's response to Bulletin Item No. 10 is acceptable.

11. Bulletin Item No. 11 requested licensees to review their prompt reporting procedures for NRC notification to assure that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time, an open, continuous communication channel shall be established and maintained with the NRC.

The existing Turkey Point notification procedures were revised to specify that the NRC be notified within one hour of the time the reactor is not in a controlled or expected condition of operating. Provisions are included for establishing and maintaining a continuous open channel of communication with the NRC using the dedicated telephone line established for this purpose. In addition, the actions specified by Bulletin Item No. 11 have been incorporated into the requirements of Section 50.72 of 10 CFR Part 50, effective immediately upon issuance on February 29, 1980.

We find the licensee's action in response to Bulletin Item No. 11 acceptable.

12. In Bulletin Item No. 12, licensees were requested to review operating modes and procedures to deal with significant amounts of hydrogen gas that may be generated during a transient or other accident that would either remain inside the primary system, or be released to the containment.

In response to this bulletin item, the licensee reviewed the existing Turkey Point procedures regarding removal of hydrogen gas from the containment using the two recombiners, purge blowers, and associated analyzers and piping provided for this purpose. This review emphasized

the accessibility, shielding, operability, sampling, and maintenance of the recombiner system. The licensee also made a comprehensive and knowledgeable evaluation of how hydrogen gas may be dealt with in the primary coolant system. The various methods which could be used for dealing with hydrogen in the primary coolant system were described. A procedure was prepared and implemented in 1979.

In addition to the above action, the licensee also participated, as a member of the Westinghouse Owners Group, in the effort to develop generic guidelines for emergency operating procedures in response to Item 2.1.9 of NUREG-0578. Included in this effort was the consideration of the treatment of non-condensable gas in the primary coolant system. Our evaluation of Item 2.1.9 implementation will be reported in a separate document.

Based on our review, we find that the licensee has provided an adequate response to Bulletin Item No. 12.

13. Bulletin Item No. 13 requested licensees to propose changes, as required, to those plant Technical Specifications which had to be modified as a result of implementing Bulletin Action Item Nos. 1 through 12, and to identify design changes necessary in order to effect long-term resolution of these items.

In its May 2, 1979 application, the licensee identified the one change to the Turkey Point Technical Specifications necessitated by actions required by this bulletin. This change was required to implement two-out-of-three low-low Pressurizer Pressure Safety Injection actuation (see Bulletin Item No. 3).

We find the licensee's response to Bulletin Item No. 13 acceptable.

CONCLUSIONS

Based on our review of the information provided by the licensee, we conclude that the licensee has correctly interpreted IE Bulletins 79-06A and 70-06A, Revision 1. The actions taken demonstrate the licensee's understanding of the concerns arising from the Three Mile Island, Unit

No. 2 accident in relation to their implications on its own operations, and provide added assurance for the protection of the public health and safety during plant operation.