ENCLOSURE NO. 1

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

March 25, 1980

Docket No.: 50-338

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric & Power Company
P. O. Box 26666
Richmond, Virginia 23261

Dear Mr. Ferguson:

SUBJECT: NRC STAFF EVALUATION OF VEPCO RESPONSES TO IE BULLETINS 79-06A AND 79-06A, REVISION 1, FOR NORTH ANNA POWER STATION, UNIT NO. 1

We have reviewed the information provided by your letters dated April 26 and June 29, 1979 in response to IE Bulletins 79-06A and 79-06A, Revision 1 for North Anna Power Station, Unit No. 1. We have also reviewed your October 15, 1979 letter which responded to our August 23, 1979 letter requesting additional information regarding the aforementioned bulletins. 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 WCAP-9600, respectively) are documented in NUREG-0623 and NUREG-0611, respectively. You will be kept informed regarding the requirements for the North Anna Unit 1 plant resulting from these reviews by separate correspondence.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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

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

NORTH ANNA POWER STATION, UNIT NO. 1
DOCKET NO. 50-338

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 (Revision 1), respectively, to Virginia Electric and Power 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 26, 1979, the licensee provided its response to the aforementioned bulletins for North Anna Power Station, Unit 1 (North Anna 1). The licensee supplemented its response by letter dated June 29, 1979, providing clarification and elaboration of certain of the Bulletin Action Items in response to our expressed concerns. Following our review of the two licensee submittals, we requested additional information regarding the licensee's responses in our August 23, 1979 letter. By letter dated October 15, 1979, the licensee provided the requested information. Our evaluation of the licensee's responses, as supplemented, is provided below.

EVALUATION

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

1. In Bulletin Action 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.

On April 21, 1979, 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 Section Leader, an Operator Licensing Branch (OLB/NRR) representative, and the facility Principal/Resident Inspector. Attendance was documented and the briefing was videotaped for later presentation to any absentees at a later briefing by the NRC 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 NRC briefing to be an acceptable response to Bulletin Action Item No. 1.

2. Action Item 2 of the Bulletin 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)

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Westinghouse Owners Group Representatives

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(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.

On April 21, 1979, 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 Section Leader, an Operator Licensing Branch (OLB/NRR) representative, and the facility Principal/Resident Inspector. Attendance was documented and the briefing was videotaped for later presentation to any absentees at a briefing by the NRC 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 NRC briefing to be an acceptable response to Bulletin Action Item No. 1.

2. Action Item 2 of the Bulletin 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.

In its October 15, 1979 supplemental response, the licensee stated that a chart with saturation and 50 Fahrenheit degrees subcooling curves has been placed in the Control Room. Training of operators on the natural circulation mode of operation has been carried out and documented. Also, an engineering review has been conducted to determine a mechanism which will warn the operator that he is losing the margin to saturation. This method would provide the operator with the ability to trend this information. These activities represent part of the licensee's response to the requirements of Item 2.1.3.b of NUREG-0578.

The licensee also identified the instrumentation which is currently available to the operator for recognition of void formation and to determine whether core cooling is being achieved by the natural circulation mode in the event of total loss of forced reactor coolant flow.

The licensee has changed the plant emergency procedure regarding loss of reactor coolant flow to provide the operator with the indication and actions required to establish and maintain natural circulation in case that total forced reactor coolant flow is lost.

The emergency procedures dealing with a LOCA, loss of secondary coolant, and deteriorating pressure conditions were changed to incorporate the reactor coolant pump trip requirements specified by IE Bulletin 79-06C.

The licensee revised the emergency procedure for loss of reactor coolant flow to provide the operator with guidance to enhance core cooling by natural circulation. This procedure instructs the operator on methods to be used in feeding and bleeding the steam generators and the instruments to be used to verify that core cooling by natural circulation has been established.

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 satisfy NUREG-0578 requirements, the licensee should have incorporated the guidelines into the North Anna 1 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 Action Item No. 2.

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

3. Bulletin Action 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.

In its June 29, 1979 response, the licensee stated that the pressurizer level bistables which input to safety injection initiation had been placed in the trip mode using an Abnormal Procedure (AP). Trip status lights on the control board confirm that the action has been completed. Subsequently, in July 1979, operating procedures were revised to include verification that these bistables were in the trip mode before placing the plant in operation. A standing order was issued requiring operators to manually initiate safety injection when the primary system pressure was below the actuation setpoint. On December 28, 1979, we issued Amendment No. 16 to the North Anna 1 operating license. This license amendment 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 two-out-of-three low pressurizer pressure condition regardless of the pressurizer level. We consider the licensee's response to Bulletin Action Item No. 3 acceptable.

4. Bulletin Action 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.

The North Anna 1 design provides for automatic initiation of containment isolation upon safety injection actuation, as called for in the bulletin. This aspect of the licensee's response is therefore acceptable.

Containment isolation consists of a Phase A and a Phase B isolation. Phase A involves closure of automatic valves in all non-essential process lines; Phase B isolates all remaining process lines, except for those related to engineered safety features. The reactor coolant pump seal water return line is isolated upon a Phase A signal. The seal water supply is not provided with isolation valves. The component cooling water supply and return lines for the reactor coolant pumps are isolated by a Phase B signal. The reactor coolant pumps do not trip automatically on either isolation signal. Therefore, the pumps must be manually tripped following a Phase B isolation, since component cooling water to the motor coolers and thermal barriers is lost.

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

5. In Bulletin Action Item No. 5, licensees with facilities at which the auxiliary feedwate system is not automatically initiated were requested to prepare and implement immediatel 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 at North Anna 1 is automatically initiated, with no operator action required in order to ensure adequate flow. Therefore, Bulletin Action Item No. 5 does not apply to this plant.

6. Bulletin Action 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 North Anna 1 procedures and determined that no changes or revisions were needed to comply with Bulletin Action Item No. 6.a.

In response to Bulletin Action Item No. 6.b. the licensee issued a Standing Order to the operators to ensure compliance with the requirements. In May 1979, the plant procedures were revised to implement Bulletin Action Item No. 6.b. Based on our review, we find that the licensee's response to Bulletin Action Item No. 6 is acceptable.

7. In Bulletin Action 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 mus 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.

STET In response to Bulletin Action Item No. 7.a, the licensee revised the applicable ~~North Anna~~ Beaver Valley 1 plant procedures in ~~May~~ April July 1979 to prohibit overriding engineered safety features unless continued operation of engineered safety features would result in unsafe conditions. This constitutes an acceptable response to Bulletin Action Item No. 7.a.

In response to Bulletin Action 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

50°F

(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.

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

Our Office of Inspection and Enforcement will verify that the approved Westinghouse generic safety injection termination criteria have been properly incorporated in the Beaver Valley 1 ~~North Anna 1~~ plant procedures. Pending such verification, we find that the licensee's actions with regard to this bulletin action item 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 Action Item No. 7.c). On July 26, 1979, IE Bulletin 79-06C superseded Action 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 1350 psi 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 ~~North Anna 1~~ plant procedures, we find the licensee's response to Bulletin Action Item No. 7.c acceptable. Beaver Valley 1.

In response to Bulletin Action Item No. 7.d, the licensee issued a Standing Order to North Anna 1 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 action item was incorporated in the licensee's operator training program. In its June 29, 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 Action Item No. 7.d.

8. Bulletin Action 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 response, the licensee stated that the required review was conducted by reviewing valve positions concurrently with the procedures that check or manipulate the valves. In its October 15, 1979 supplemental response, the licensee added that valve lineups on safety-related systems are completed after every refueling. Locked valves on safety-related systems are verified and documented with respect to their proper position. Safety-related valves that have position indication in the control room are verified to be in their proper positions on a shift turnover check list which has been implemented to meet the requirements of Item 2.2.1.c of NUREG-0578, "Shift and Relief Turnover Procedures."

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

9. In Bulletin Action 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.

Insert 13D

In its October 15, 1979 supplemental response, the licensee listed all systems which are designed to transfer potentially radioactive fluids from containment, indicated those systems for which high radiation interlocks exist, and identified the means by which the operability of each system listed is assured. Information pertaining to the resetting of ESFs and its effect on valves controlling the release of radioactivity was provided in the licensee's October 24, 1979 response to Item 2.1.4 of NUREG-0578. In brief, once Phase A Containment Isolation has been initiated by a safety injection signal, the automatic isolation valves can be opened only upon manual reset of the actuating signal and deliberate remote manual operation of the individual valve.

We find that the licensee has adequately addressed the concerns expressed in Bulletin Action 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 ESFs 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.

Insert p-11

In its October 15, 1979 supplemental response, the licensee provided additional information regarding this bulletin action item. The North Anna 1 Technical Specifications specify the surveillance requirements that must be completed to confirm the operability of safety-related systems. A subsystem or equipment is removed from service for preventive or corrective maintenance according to maintenance operating procedures. When a subsystem fails or is removed from service, this event is entered in an Action Statement Log to ensure that Technical Specification requirements are met. When maintenance has been completed, the controlling procedure ensures that testing of the subsystem/equipment is performed to determine operability.

Maintenance operating procedures will test the redundant subsystem/train before removal of a portion of the other subsystem/train if it does not isolate it from performing its safety function while testing. In the case of subsystems which are made inoperable for testing, it is verified that the redundant train of the system to be removed from service is not listed in the Action Statement Log, and that it has passed its last scheduled periodic test. The redundant train is visually inspected and its power supply is verified as being operable and not listed in the Action Statement log. These steps are taken and documented in the maintenance operating procedure before the system is removed from service.

Operability of a redundant emergency diesel is tested by the maintenance operation procedure in accordance with the Technical Specifications. The procedures require verification that safety-related systems powered from the redundant diesel are operable from a review of the Action Statement Log.

The licensee conducted a detailed review of periodic tests to ensure the operability of a system is determined when equipment is returned to service following testing. This review also identified equipment which is made inoperable for testing purpose.

The transfer of information about the status of safety-related systems at shift change will be accomplished according to the requirements of Item 2.2.1.c of NUREG-0578.

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

11. Bulletin Action 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 North Anna 1 notification procedures were revised on April 30, 1979 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. These reporting requirements have been posted on a bakelite sign within view of the Shift Supervisor's desk. We find the licensee's action in response to Bulletin Action Item No. 11 acceptable.

12. In Action 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 action item, the licensee reviewed the existing North Anna 1 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.

In addition, in its October 15, 1979 supplemental response, the licensee identified the various methods covered by existing procedures for removing hydrogen gas from the reactor coolant system.

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

13. This bulletin action item 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 October 15, 1979 supplemental response, the licensee identified the one change to the North Anna 1 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 (from Bulletin Action Item No. 3).

We find the licensee's response to Bulletin Action 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 79-06A, 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.

ENCLOSURE NO. 2

STATUS OF REVIEW OF RESPONSES TO IE BULLETINS 79-06A & 79-06A (REVISION 1)

(All dates are in 1979)

Plant Name	Date of Licensees' Responses	Date of Draft BEVR	Date of Request for Information	Date of IE Inspection Report/Comments
Beaver Valley 1	4/30, 5/14, 5/17, 7/12, 9/21	6/7	9/13	8/10 (report)
D. C. Cook 1&2	5/1, 6/6, 6/25, 7/25, 10/11	6/7	9/7	6/26 (report)
Farley 1	4/24, 6/22	6/7	N/A	6/22 (comments)
Ginna **	4/28, 6/22	6/7	N/A	
Haddam Neck *	4/24, 5/14, 5/18, 5/31, 6/26, 7/6	6/8	N/A	
Indian Pt. 2	4/26, 6/22	6/7	N/A	
Indian Pt. 3	4/26, 6/6, 6/20	6/7	N/A	
Kewaunee	4/30, 6/20, 7/13	6/7	N/A	
North Anna 1 *	4/26, 6/29, 10/15	6/7	8/23	5/15, 6/8, 6/25 (reports) 6/22 (comments)
Point Beach 1&2	4/19, 4/27, 8/23, 9/20, 9/26	6/7	8/3	
Prairie Island 1 & 2	4/30, 5/18, 6/22	6/7	N/A	
H.B. Robinson	4/23, 6/28, 7/12, 8/28	6/7	8/9	6/22 (comments)
Salem 1 *	4/25, 5/11, 6/1, 7/13, 8/14	6/7	N/A	
San Onofre 1	4/19, 5/3, 5/23, 6/25, 8/28	6/8	8/7	6/1, 6/27 (reports)

ENCLOSURE NO. 2 - continued - page 2

(All dates are in 1979)

Plant Name	Date of Licensees' Responses	Date of Draft BEVR	Date of Request for Information	Date of IE Inspection Report/Comments
Surry 1&2	4/26,6/26, 7/6 (Clarification of certain previous responses furnished to ORPM by telecopy)	6/7		6/12 (report) 6/22 (comments)
Trojan	4/24,5/4, 5/18, 6/25,7/17,9/14	6/7	N/A	5/16 (report) 6/27 (comments)
Turkey Point 3&4	4/24,6/18, 6/25	6/7	N/A	6/26 (comments)
Yankee Rowe	4/26,5/16,5/24, 6/20, 8/30	6/7	8/7	
Zion 1&2	4/27,5/17, 6/22	6/7	N/A	6/14 (report)

* Completed

** BEVR already updated, draft at CRESS for typing.