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

August 6, 1980

Docket No. 50-29

Mr. James A. Kay
Senior Engineer-Licensing
Yankee Atomic Electric Company
25 Research Drive
Westborough, Massachusetts 01581

Dear Mr. Kay:

SUBJECT: NRC STAFF EVALUATION OF YAECO RESPONSES TO IE BULLETINS 79-06A AND 79-06A, REVISION 1, FOR YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)

We have reviewed the information provided by your letters dated April 26, May 16, May 18, and June 20, 1979 in response to IE Bulletins 79-06A and 79-06A, Revision 1 for the Yankee-Rowe plant. We have also reviewed your August 30, 1979 letter which responded to our August 7, 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 intent of said 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 79-06A (Westinghouse reports WCAP-9584 and WCAP-9600, respectively) are documented in NUREG 0623 and NUREG 0611, respectively. You will be informed regarding the requirements for the Yankee-Rowe plant resulting from these reviews by separate correspondence.

Sincerely,

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
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Enclosure and cc:
See page 2

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EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETINS

79-06A AND 79-06A (REVISION 1)

YANKEE NUCLEAR POWER STATION (YANKEE-ROWE)
DOCKET NO. 50-29

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 Yankee Atomic Electric 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 responses to the aforementioned bulletins for Yankee-Rowe. The licensee supplemented its response by letters dated May 16, May 24, and June 20, 1979 providing clarification and elaboration of certain of the Bulletin Action Items in response to our expressed concerns. 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.

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. Attendance was documented and a briefing was given to any absentees at a later date 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 August 30, 1979 supplemental response, the licensee stated that even in a partially voided condition natural circulation cooling will remain effective since Yankee-Rowe has: (1) an adequate heat sink in the steam generators either through the normal or the emergency feedwater systems, and (2) the capability to verify the maintenance of a flow path by examination of several key parameters including reactor coolant temperatures in the reactor core and in the reactor cooling loops.

   The licensee has emergency procedures which provide the operator with guidance for maintaining the heat sink and flow path necessary for natural circulation under reactor coolant conditions above the saturation pressure at the core outlet or in extreme cases where voids may be present.

   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 Yankee-Rowe 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.

   Yankee-Rowe does not use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection. Emergency operating procedures currently instruct the operators to immediately verify that safety injection actuation has occurred as soon as RCS pressure has decreased below the actuation setpoint.

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 Yankee-Rowe design provides for all lines having automatic trip valves to be isolated simultaneously on a containment isolation actuation signal of 5 psig (Refer to Technical Specification Table 3.6-1 for a complete listing of these lines). The Yankee Rowe emergency operating procedures have been revised to instruct the operator to manually initiate a limited containment isolation following a safety injection actuation due to a low pressure condition. This manual initiation does not include two valves, TV-205 (component cooling from Reactor Coolant Pumps) and TV-405 (auxiliary steam to the Emergency Boiler Feed Pump). These valves are not tripped to maintain operability of the Reactor Coolant Pumps and the Emergency Boiler Feed Pump.

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

In addition to the above responses, our review of the licensee's responses to Category "A" Lessons Learned, Item 2.1.4 "Containment Isolation" provides further assurance that our concerns with containment isolation at Yankee-Rowe have been satisfied.

5. In Bulletin Action 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.

   An additional operator with no other assigned concurrent duties was added to each operating shift as of April 12, 1979. The operator utilizes a recently installed redundant communications system and is required to initiate adequate feedwater to the steam generator for those transients or accidents the consequence of which can be limited by such actions.

   We find the licensee's response to Bulletin Action Item No. 5 acceptable.

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 has listed the relevant plant indications that would enable an operator to determine if the power-operated relief valve (PORV) is open. The licensee has stated that the PORV can be isolated by a normally open motor operated isolation valve. This complies with Bulletin Action Item No. 6.a.

   In its June 20, 1979 supplemental response, the licensee stated that on May 11, 1979 a Special Operating Order was issued to the operating personnel to ensure compliance with the required actions. The formal procedure changes were approved on May 14, 1979. All operators have been trained in these changes during training sessions held on May 17 and 18, 1979. This implements Bulletin Action Item No. 6.b.

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 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 consideration 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.

In response to Bulletin Action Item No. 7.a, the licensee has reviewed the operating procedures and training instructions to ensure that the operators will not override automatic actions of engineered safety features, unless their continued operation will result in unsafe plant operation. During training sessions on May 18 and 19, 1979, this concern was stressed to the licensed operators.

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

(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-0511) 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 Yankee Rowe 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

September 14, 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 Yankee Rowe plant procedures, we find the licensee's response to Bulletin Action Item No. 7.c acceptable.

In response to Bulletin Action Item 7.d, the licensee issued a Special Operating Order to the operators on April 11, 1979 instructing them against overreliance on pressurizer level indication and to use other plant parameters in assessing water inventory and plant conditions.

In its June 20, 1979 supplemental response the licensee stated that Emergency Operating Procedures have been changed to assure that the operator utilizes a combination of plant parameters as indications of main coolant system conditions. Licensed operators have been instructed on the changes during training sessions on May 17 and 18, 1979.

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 its June 20, 1979 supplemental response, the licensee stated that plant procedures have been reviewed to ensure that alignments and controls for all safety related valves, necessary for proper operation of the engineered safety features, are satisfied.

   Procedures have been reviewed and revised to assure periodic surveillance of all locked safety related valves. Also, plant maintenance, testing and surveillance for engineered safety features have been reviewed and revised to ensure that they include specific "return to service" requirements.

   All licensed operators were trained in the procedure changes during training sessions on May 17 and 18, 1979.

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

9. In Bulletin Action Item No. 9, the 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 responses, the licensee identified all lines which are designed to transfer potentially radioactive fluids from containment. In its June 20, 1979 supplemental response, the licensee stated that procedures revisions are in progress to assure prior to resetting of engineered safety features, that valves closed to prevent a transfer of potentially radioactive fluids from containment, remain closed; also, to assure alignment of the containment isolation system so that any reset of engineered safety features will not cause the opening of trip valves which have been closed. Inspection and Enforcement has verified that acceptable procedure revisions have been completed.

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

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.

In its August 30, 1979 supplemental response, the licensee stated that the maintenance and test procedures were revised where necessary to provide for verification by the shift supervisor that redundant systems are operable before a safety related system is taken out of service. The licensee provided the details on these procedure revisions which include notification of the shift supervisor following return to service of a safety related system, that all maintenance and testing had been completed; requirements of identified tests to be performed under the direction of the shift supervisor; and requirements for the shift supervisor to sign off on the applicable procedure before declaring a system operable.

The licensee also stated that information on the status of safety related systems is passed on from shift to shift on shift turnover in accordance with the shift turnover procedure. This procedure requires that all previously unreviewed entries and active entries in the identified plant logs be reviewed by the shift supervisors and the control room operators. In addition, when the shift supervisor and the control room operators were being relieved from duty, they are required to orally communicate a detailed plant summary of past and present plant status.

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.

    In its June 20, 1979 supplemental response, the licensee stated that the installation of one of two proposed direct and dedicated telephone lines between Yankee-Rowe and the NRC headquarters has been completed. This continuous open channel of communication has been established for promptly notifying the NRC of operational conditions. The licensee also stated that the second line, for communicating radiological and environmental information during an emergency was scheduled to be installed by September 1, 1979.

    The licensee has issued a plant operational memo which instructs the operators on the use of this communications system including the requirement of notifying the NRC within one hour from the time when the reactor is not in a controlled or expected condition.

The actions specified in Bulletin Action Item No. 11 have subsequently been incorporated in the requirements of Section 50.72 in 10 CFR Part 50 which became effective immediately upon issuance on February 29, 1980.

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 its April 26, 1979 response, the licensee has made a knowledgeable evaluation of operating modes for dealing with hydrogen gas that may be generated during a transient in other accidents both inside the primary system and in the containment. He states that the various actions necessary to mitigate the effect of hydrogen in the primary coolant system will be incorporated in the operating procedures, and that curent procedures adequately address the removal of hydrogen from the containment.

    In its June 20, 1979 supplemental response, the licensee stated that an emergency operating procedure, addressing the control of hydrogen gas in the main coolant system has been revised. The licensee also stated that all licensed operators have been trained in these changes during training sessions on May 17 and 18, 1979.

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

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.

This conclusion notwithstanding, it should be recognized that further actions will result from the staff's review of operating plants using nuclear steam supply systems designed by Westinghouse (documented in NUREG-0611). Additional changes have resulted from the staff's implementation of the requirements contained in NUREG-0578 (e.g., the actions taken in response to Action Item 6 of Bulletin 79-06A regarding the PORVs). Our evaluation of such matters will be provided in other documents.