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

December 18, 1979

Docket No. 50-293

Mr. G. Carl Andognini
M/C NUCLEAR
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Mr. G. Carl Andognini:

SUBJECT: NRC STAFF EVALUATION OF BECO RESPONSES TO IE BULLETIN 79-08 FOR PILGRIM NUCLEAR POWER STATION UNIT 1

We have completed our review of the information that you provided in your letters dated April 25 and May 15, 1979 in response to IE Bulletin 79-08 for the Pilgrim Nuclear Power Station, Unit No. 1. We have also completed our review of the supplemental information that you provided in your letters of August 21, 1979 and November 7, 1979.

We have concluded that you have taken the appropriate actions to meet the requirements of each of the eleven action items identified in IE Bulletin 79-08. A copy of our evaluation is enclosed.

As you know, NRC staff review of the Three Mile Island, Unit 2 (TMI-2) accident is continuing and other corrective actions may be required at a later date. For example, the Bulletins and Orders Task Force is conducting a generic review of operating boiling water reactor plants. Specific requirements for your facility that result from these and other TMI-2 investigations will be addressed to you in separate correspondence.

Sincerely,

Thomas A. Ippolito, Chief
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Enclosure:
NRC Staff Evaluation

cc w/enclosure:
See next page

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EVALUATION OF LICENSEE'S RESPONSES

TO

IE BULLETIN 79-08

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-293

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Introduction

By letter dated April 14, 1979, we transmitted Office of Inspection and Enforcement (IE) Bulletin 79-08 to Boston Edison Company (BECo or the licensee). IE Bulletin 79-08 specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979. By letter dated April 25, 1979, BECo provided responses to Action Items 1 through 10 of IE Bulletin 79-08 for the Pilgrim Nuclear Power Station, Unit 1. BECo supplemented this response by a letter dated May 15, 1979 to provide the response to Action Item 11 of IE Bulletin 79-08.

The NRC staff review of the BECo responses led to the issuance of requests for additional information regarding the BECo responses to certain action items of IE Bulletin 79-08. These requests were contained in a letter dated July 20, 1979. By letters dated August 21 and November 7, 1979, BECo responded to the staff's requests for additional information.

The BECo responses to IE Bulletin 79-08 provided the basis for our evaluation presented below. In addition, the actions taken by the licensee in response to the bulletin requirements and subsequent NRC requests were verified through onsite inspections by IE inspectors.

Evaluation

Each of the 11 action items requested by IE Bulletin 79-08 is repeated below followed by our criteria for evaluating the response, a summary of the licensee's response and our evaluation of the response.

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 March 28, 1979 accident included in Enclosure 1 to IE Bulletin 79-05A.

    a. This review should be directed toward understanding: (1) the extreme seriousness and consequences of the simultaneous blocking of both trains of a safety system 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; and (3) 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 5a of this bulletin); 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 shall participate in this review and such participation shall be documented in plant records.

The licensee's response was evaluated to determine that (1) the scope of review was adequate, (2) operational personnel were properly instructed and (3) personnel participation in the review was documented in plant records.

The licensee's response dated April 25, 1979 committed to a training program to be conducted under formal classroom conditions addressing each of the suggested items. This review is to be documented in the training records. A supplemental response dated August 21, 1979 reported that the required actions had been completed for all licensed operators and plant management on May 18, 1979.

We conclude that the licensee's scope of review, instructions to operating personnel and documented participation satisfies the intent of IE Bulletin 79-08, Item 1.

2. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to initiate containment isolation, whether manual or automatic, of all lines whose isolation does not degrade needed safety features or cooling capability, upon automatic initiation of safety injection.

The licensee's response was evaluated to verify that containment isolation initiation design and procedures had been reviewed to assure that (1) manual or automatic initiation of containment isolation occurs on automatic initiation of safety injection and (2) all lines (including those designed to transfer radioactive gases or liquids) whose isolation does not degrade cooling capability or needed safety features were addressed.

The licensee's April 25, 1979 response noted that a review of the containment isolation which occurs in connection with initiation of safety injection was conducted. The following lines were identified that did not receive automatic isolation upon safety injection:

(1) Reactor Building Component Cooling Water

(2) Instrument Air/Nitrogen Supply

(3) Instrumentation Lines

(4) RHR to Spent Fuel Pool Demineralizer

(5) Hydrogen Analyzer

(6) Reactor Building to Torus Vacuum Breakers

(7) Torus Make-up

(8) Main Steam

(9) Main Steam Drain

(10) Reactor Water Sample

The licensee committed to revise its procedures for Items (1), (2), (4) and (7) to provide for manual isolation. We concur with the licensee's bases for

not requiring isolation for Items (3), (5), (6) and (8) upon safey injection. In a supplemental response dated August 21, 1979, the licensee reported that the appropriate procedures had been revised and were in place by June 20, 1979. In addition, the licensee committed to modify the main steam line drain and reactor water sample lines by the end of the 1980 refueling outage to provide isolation upon high drywell pressure.

We conclude that the licensee's review of containment isolation initiation design and procedures satisfy the intent of IE Bulletin 79-08, Item 2.

3. Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense.

The licensee's response was reviewed to assure that (1) it described the automatic and manual actions necessary for the proper functioning of the auxiliary heat removal systems when the main feedwater system is not operable and (2) the procedures for any necessary manual actions were described in summary form.

The licensee's response dated April 25, 1979 stated that the high pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system function as auxiliary heat removal systems when the main feedwater system is inoperable. The automatic and manual actions necessary for the proper functioning of HPCI and RCIC were described in summary form. We acknowledge the capability of these systems to provide the required heat removal action.

We conclude that the licensee's procedural summary of automatic/manual actions necessary for the proper functioning of auxiliary heat removal systems used when the main feedwater system is inoperable satisfies the intent of IE Bulletin 79-08, Item 3.

4. Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems.

The licensee's response was evaluated to determine that (1) all uses and types of vessel level indication for both automatic and manual initiation of safety systems were addressed, (2) it addressed other instrumentation available to the operator to determine changes in reactor coolant inventory and (3) operators were instructed to utilize other available information to initiate safety systems.

The licensee's April 25, 1979 response included a listing of all reactor vessel level instrumentation in use at the plant. The types of vessel level instrumentation in use at the plant included level indicating switches, transmitters, meters and recorders. The indicated ranges vary from 205 to 797 inches and are available both in the control room and the reactor building. In the supplemental response dated August 21, 1979, the licensee provided a listing of other instrumentation which would be available to the operator to determine changes in reactor coolant inventory. The supplemental response further committed to complete operator training on the use of all available instrumentation in this area by October 9, 1979.

We conclude that the licensee's description of the uses and types of reactor vessel level/inventory instrumentation and instructions to operators regarding the use of this information satisfies the intent of IE Bulletin 79-08, Item 4.

5. Review the actions 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 will result in unsafe plant conditions (e.g., vessel integrity).

   b. Operators are provided additional information and instructions to not rely upon vessel level indication alone for manual actions, but to also examine other plant parameter indications in evalating plant conditions.

The licensee's response was evaluated to determine that (1) it addressed the matter of operators improperly overriding the automatic actions of engineered safety features, (2) it addressed providing operators with additional information and instructions to not rely upon vessel level indication alone for manual actions and (3) that the review included operating procedures and training instructions.

The licensee stated in the April 25, 1979 response that procedural changes were being initiated to provide more explicit instructions to operators regarding overriding automatic actions of engineered safety features and control of reactor vessel water level. The supplemental response dated August 21, 1979 confirmed that all available instrumentation for control of reactor vessel level would be employed and procedural revisions would be complete by October 1, 1979.

We conclude that the licensee's review of operating procedures and training instructions satisfies the intent of IE Bulletin 79-08, Item 5.

6. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system start-up, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

The licensee's response was evaluated to assure that (1) safety-related valve positioning requirements were reviewed for correctness, (2) safety-related valves were verified to be in the correct position and (3) positive controls were in existence to maintain proper valve position during normal operation as well as during surveillance testing and maintenance.

The licensee's response dated April 25, 1979 described the methods used to verify safety-related valve positions during normal operation, maintenance and surveillance testing. In the supplemental response dated August 21, 1979, the

licensee committed to review all station operating procedures to verify proper safety-related valve positioning requirements by October 1, 1979. The supplement further confirmed that both inaccessible and accessible safety-related valve positions had been verified correct as required.

We conclude that the licensee's review of safety-related valve positioning requirements, valve positions and positive controls to maintain proper valve positions satisfies the intent of IE Bulletin 79-08, Item 6.

7. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping, venting or other release of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

   c. The basis on which continued operability of the above features is assured.

The licensee's response was evaluated to determine that (1) it addressed all systems designed to transfer potentially radioactive gases and liquids out of primary containment, (2) inadvertent releases do not occur on resetting engineered safety features instrumentation, (3) it addressed the existence of interlocks, (4) the systems are isolated on the containment isolation signal, (5) the basis for continued operability of the features was addressed and (6) a review of the procedures was performed.

In the April 25, 1979 response, the licensee reported that there were two liquid systems and two gas systems for transferring potentially radioactive materials from the primary containment. The two liquid systems are (1) containment sumps, and (2) the residual heat removal (RHR) system to radwaste.

Both of these systems isolate on a containment isolation signal (CIS). The isolation cannot be overridden while the CIS is present. Upon reset, the RHR system to radwaste isolation remains intact; however, a potential exists for the sump isolation valves to open upon CIS reset, thus setting the stage for an inadvertent liquid release. In the supplemental response dated August 21, 1979, the licensee described a modification to the sump pump control logic to preclude inadvertent pump/valve operation upon resetting CIS. The modification has been installed and is operational.

The two gas systems are: (1) the containment atmospheric control (CAC) system and (2) the stand-by gas treatment system (SGTS). The CAC system will isolate on a CIS. The SGTS activates to maintain a negative pressure in the secondary containment. The CAC system has override capability via individual keylock switches to allow venting the primary containment via the SGTS when a CIS is present. CAC system valves A05041A and A05041B are normally open to maintain torus/drywell differential pressure and close automatically in the event of a CIS. Upon CIS reset, these two series valves would open a two-inch line from the torus to the SGTS, thus setting the stage for an inadvertent gaseous release. In the supplemental response dated November 7, 1979, the licensee committed to revise its procedures to preclude an inadvertent opening of these CAC system valves upon CIS reset.

We conclude that the licensee's review of systems designed to transfer radioactive gases and liquids out of primary containment to assure that undesired pumping, venting, or other release of radioactive liquids and gases will not occur satisfies the intent of IE Bulletin 79-08, Item 7.

8. Review and modify as necessary your maintenance and test procedures to ensure that they require:

   a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

   b. Verification of the operability of safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

The licensee's response was evaluated to determine that operability of redundant safety-related systems is verified prior to the removal of any safety-related system from service. Where operability verification appeared only to rely on previous surveillance testing within Technical Specification intervals, we asked that operability be further verified by at least a visual check of the system status to the extent practicable, prior to removing the redundant equipment from service. The response was also evaluated to assure provisions were adequate to verify operability of safety-related systems when they are returned to service following maintenance or testing. We also checked to see that all involved reactor operational personnel in the oncoming shift are explicitly notified during shift turnover about the status of systems removed from or returned to service since their previous shift.

The licensee's response dated April 25, 1979 suggested that the operability of redundant safety-related systems might be verified through reliance on Technical Specification periodic tests. A supplemental response dated August 21, 1979 stated that Pilgrim Station procedures require that operability tests be performed on redundant safety-related systems immediately prior to removal of any safety-related system from service.

We conclude that the licensee's review and modification of maintenance, test and administrative procedures to assure the availability of safety-related systems and operational personnel knowledge of system status satisfies the intent of IE Bulletin 79-08, Item 8.

9. Review your prompt reporting procedures for NRC notification to assure that 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 NRC.

The licensee's response was evaluated to determine that (1) prompt reporting procedures required or were to be modified to require that the NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation and (2) procedures required or were to be modified to require the establishment and maintenance of an open continuous communication channel with the NRC following such events.

The licensee's response dated April 25, 1979 reported that procedural changes would be necessary to provide the required communications. The supplemental response dated August 21, 1979 confirmed that all station reporting procedures had been revised as necessary to implement the reporting guidelines. In addition, the station emergency procedures were similarly revised on or before August 15, 1979.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 9.

10. 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.

The licensee's response was evaluated to determine if it described the means or systems available to remove hydrogen from the primary system as well as the treatment and control of hydrogen in the containment.

The licensee stated in the April 25, 1979 response that the emergency procedure for post-accident venting had been reviewed and determined to be adequate to control hydrogen released to the containment. In the supplemental response dated August 21, 1979, the licensee reported that a new procedure, "Loss of Coolant with No Pipe Breaks," had been developed to remove hydrogen from the primary system.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 10.

11. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the items above.

The licensee's response was evaluated to determine that a review of the Technical Specifications had been made to determine if any changes were required as a result of implementing Items 1 though 10 of IE Bulletin 79-08.

The licensee reported in its letter dated May 15, 1979 that no Technical Specification changes were required as a result of implementing Items 1 through 10 of IE Bulletin 79-08.

We conclude that the licensee's response satisfies the intent of IE Bulletin 79-08, Item 11.

Conclusion

Based on our review of the information provided by the licensee to date, we conclude that the licensee has correctly interpreted IE Bulletin 79-08. The actions taken demonstrate the licensee's understanding of the concerns arising from the TMI-2 accident in reviewing their implementation on Pilgrim 1 operations, and provide added assurance for the protection of the public health and safety during the operation of Pilgrim 1.

References

1. IE Bulletin 79-05, dated April 1, 1979.

2. IE Bulletin 79-05A, dated April 5, 1979.

3. IE Bulletin 79-08, dated April 14, 1979.

4. BECo letter, #79-79 dated April 25, 1979.

5. BECo letter, #79-93 dated May 15, 1979.

6. NRC staff letter, T. Ippolito to G. Andognini, dated July 20, 1979.

7. BECo letter, #79-165 dated August 21, 1979.

8. BECo letter, #79-229 dated November 7, 1979.

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