SAFETY EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETIN 79-08

NORTHEAST NUCLEAR ENERGY COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

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Introduction

By letter dated April 14, 1979, we transmitted IE Bulletin No. 79-08 to Northeast Nuclear Energy Company (NNECO or the licensee). This Bulletin 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 24, 1979, NNECO provided responses to action items one through 10 of IE Bulletin 79-08 for the Millstone Nuclear Power Station, Unit No. 1 (Millstone-1). NNECO supplemented this response, by letters dated May 14, 1979 and June 15, 1979, to provide its response to action item 11 of IEB 79-08 and to clarify and elaborate on certain of the items as a result of discussions with the NRC staff.

The NNECO responses to IE Bulletin 79-08 provided the basis for our evaluation presented below.

Evaluation

Each of the 11 action items requested by IE Bulletin 79-08 is repeated below followed by a summary of the licensee 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 3/28/79 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.

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The licensee has reported that a lesson plan responsive to items 1a and 1b has been prepared and presented to all licensed and unlicensed operators as well as plant management and supervisors with operational responsibility.

On the basis of the NNECO response, we conclude that the intent of IE Bulletin 79-08 items 1a, b and c has been satisfied.

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 in its April 24, 1979 letter reported that the primary containment isolation design has been reviewed and it has been confirmed by this review that the required containment isolation does occur in parallel with the automatic initiation of any of the safety injection systems. This is because containment isolation and safety injection utilize the same water level sensors. The response also stated that the boiling water reactor design provides containment and reactor coolant pressure boundary isolation (excluding emergency core cooling and make-up systems) and that the isolation occurs upon reactor vessel low water level or high drywell pressure prior to, or simultaneous with, initiation of the emergency core cooling and safety injection systems. The licensee's response stated that the isolation valves will remain closed until operator action is taken, even if the initiating signal clears. (A detailed description of the containment and system isolations can be found in the Millstone-1 Final Safety Analysis Report and they are summmarized in the Millstone-1 Technical Specifications.)

We have concluded that the review by the licensee which confirmed the adequacy of existing written procedures satisfies 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 in its April 24, 1979 letter reported that Millstone-1 utilizes the isolation condenser system as an auxiliary heat removal device when the main feedwater system is not operable. The system is designed to automatically initiate when reactor pressure reaches 1085 pounds per square inch gauge for 15 seconds. This system relies upon the natural circulation of steam from the reactor vessel through the isolation condenser and returns condensate to the vessel. Makeup for the shell side is automatic and is supplied from the station fire water or condensate transfer systems. The isolation condenser system may also be manually initiated, by opening one valve either from the control room or locally. Procedures exist for these evolutions. For long term operation in this cooling mode, the control rod drive pumps may be used to replenish the coolant lost by insignificant primary system boundary leakage.

If the isolation condenser system is not available for cooling, the plant has the ability to maintain cooling using the low pressure coolant injection system after manually depressurizing with the automatic pressure relief system. Procedures currently exist for these evolutions.

We agree that this capability exists and conclude that the licensee response 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 in its response has reported that reactor vessel water level in the boiling water reactor is continuously monitored by seven indicators or recorders for normal, transient and accident conditions. Those monitors, used to provide automatic safety equipment initiation, are arranged in a redundant array with two instruments in each of two or more independent electronic divisions. Thus, adequate information is provided to automatically initiate safety actions and provide the operator with assurance of the vessel water level at all times. In its letter dated

June 15, 1979 the licensee reported that the operating procedures reflect the requirements for the operators to also rely upon the information provided by the instrumentation discussed in the response to IE Bulletin 79-08 item 5b. These water level measurement devices have operated reliably in boiling water reactor plants for 20 years.

The range of reactor vessel water level from below the bottom of the active fuel area up to the top of the vessel is covered by a combination of narrow and wide-range instruments. Level is indicated and recorded in the control room.

A separate set of narrow-range level instrumentation on separate condensing chambers provides reactor level control via the reactor feedwater system. This set also indicates and records in the control room.

The safety-related systems or functions served by safety-related reactor water level instrumentation are:

Reactor scram
Feedwater coolant injection system
Core spray system
Low pressure coolant injection system
Automatic pressure relief system
Main steam isolation valve closure
Primary containment isolation

All systems automatically initiate on low reactor water level. The feedwater coolant injection system will control in level control mode if and when level is restored to the normal operating range. The core spray and low pressure coolant injection systems will continue to operate until manually shut down.

The licensee in its letter dated April 24, 1979 stated that in the unlikely event that vessel level indication were in doubt, the operators would continue to allow the feedwater coolant injection, core spray and low pressure coolant injection systems to operate, overflowing the vessel to the torus via the automatic pressure relief system valves. Existing procedures have been modified to clarify this operation.

On the basis of the information provided by NNECO we have concluded that the intent of IE Bulletin 79-08 item 4 has been satisfied.

5. 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 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 evaluating plant conditions.

The licensee has reported the following:

a. The Millstone-1 plant's procedures and training currently are in agreement with the NRC position on not overriding automatic safety functions.

b. Over a dozen other types of instrumentation in the boiling water reactor provide the operator with indirect indication of reactor vessel coolant inventory changes and could inform the operator of the need to take corrective actions. The licensee reported that a review of operating and emergency procedures showed that various parameters are monitored for each type of accident. Operators are required to first confirm that automatic functions have occurred. Operator actions, as required in the procedures, are based upon the monitoring of many redundant parameters, one of which is vessel water level. Some of the instrumentation which the operator can use to determine changes in reactor coolant inventory or other abnormal conditions are

Drywell high pressure
Drywell high radioactivity levels
Suppression pool high temperature
Safety relief valve discharge high temperature
High feedwater flow rates
High main steam flow
High containment and equipment area temperatures
High differential flow, reactor water clean up system
Abnormal reactor pressure
High suppression pool water level
High drywell and containment sump fill and pumpout rate

The licensee provided the following three examples of the use of this additional information by the operator. Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High safety relief valve discharge temperature would pinpoint loss of coolant via an open valve.

Other instrumentation that can signal abnormal plant status, but may not necessarily be indicative of loss of coolant are:

High neutron flux
High process monitor radiation levels
Main turbine status instrumentation
Abnormal reactor recirculation flow
High electrical current to pump motors

We have concluded on the basis of the information submitted by the licensee that the intent of IE Bulletin 79-08 item 5 has been satisfied.

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 startup, 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 confirmed by telephone on July 11, 1979 that all safety-related valve positions, positioning requirements and procedural controls which ensure that the valves have been properly positioned for operation of engineered safety features in accordance with the system design requirements have been reviewed. Where necessary, procedures have been revised to make them more inclusive. Also the administrative procedures governing surveillance testing, maintenance and system/plant startup relative to safety-related valve position verification have been reviewed. The existing procedures for surveillance testing are considered by the licensee to be adequate. The procedures for control of maintenance on safety-related equipment have been revised to specifically assure correct positioning of valves which were worked on or were used for isolation purposes. We have also confirmed with the licensee that positions of all safety-related valves, except for locked valves, are visually checked monthly. The positions of locked valves are visually checked prior to each startup and after any system manipulation that require their repositioning.

The NRC-approved Technical Specifications require that valve lineup lists be reviewed by the Plant Operations Review Committee to ensure proper valve positioning prior to operation, any time modifications are made that could affect valve lineups. Simulated or actual automatic actuation and functional system testing is also required by Technical Specifications each refueling cycle on emergency core cooling systems; core spray, low pressure coolant injection, feedwater coolant injection, isolation condenser, and automatic pressure relief. The licensee has also identified a need to develop a system to ensure that the control room drawings, including system process and instrumentation drawings are kept updated to reflect all drawing change requests, including those requests being processed.

On the basis of the information provided by the licensee, we have concluded that the intent of IE Bulletin 79-08 item 6 has been satisfied.

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 responded that systems designed to transfer potentially contaminated radioactive gases and liquids out of the primary containment include the main steam system, cleanup system, drywell equipment and floor drain systems, recirculating loop sample line, and the drywell and suppression chamber vent systems. These systems are designed to isolate on either low reactor water level or high drywell pressure. Procedurally, a sample is taken for airborne activity in the primary containment before venting. The drywell sumps are procedurally operated in manual and thus the possibility of inadvertent pumping is minimal. While no installed radiation monitoring exists for these sump systems, their discharge lines could be monitored with portable instrumentation if the potential for pumping highly contaminated water was present. The main steam system and clean up systems are equipped with process or area radiation monitors to protect against inadvertent high level releases by these paths.

   The licensee reported that each of these protective features is routinely calibrated and/or tested.

   We have concluded that the licensee response 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 all 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 reported that:

a. The administrative procedures have been revised to specify that prior to removal of safety-related systems from service the redundant systems will be verified operable. For equipment which the Technical Specifications require specific surveillance, that testing will be completed prior to removing the system from service.

b. Procedures for maintenance and testing of safety-related systems have been reviewed and changes have been made to strengthen the requirement to verify operability of safety-related systems prior to taking credit for the system(s) to satisfy Technical Specification requirements.

c. A licensed operator is required to authorize all maintenance, tests, or surveillance which affect plant systems. Prior to releasing the controlling document, the operator ensures he is aware of the effect of the activity on the system or equipment. Upon completion of the item, the document is returned to the operator for acceptance or for the purpose of returning the system to service. The administrative procedures which control these evolutions provide the required explicit notification of operational personnel whenever a safety-related system is removed from and returned to service. The control room procedures assure that during shift changes, the oncoming shift is fully informed of any abnormalities in the plant, the equipment running, and other pertinent facts about the plant status.

We have concluded that the licensee responses satisfy the intent of IE Bulletin 79-09 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 reported that a revision to the administrative procedure on communications and outside assistance has been approved. This revision incorporates the required notifications and establishment of communication channels requested in the Bulletin.

The licensee noted that the wording of the reason for immediate notification ("The reactor is not in a controlled or expected condition of operation") is general in that many different circumstances may or may not fit the definition, depending on who is interpreting the situation. Because of this the licensee requested more specific guidance on this point in order to provide more explicit instructions to the operators and duty officers. We agree that the Bulletin statement is, of necessity, a general statement and was prepared in light of our knowledge of the early sequence of events at TMI-2 prior to NRC notification. We leave it to the licensee to likewise review the TMI-2 events and, using that as guidance together with his experience in routine operations and the recognition of non-routine events, promulgate his own interpretation of prompt NRC notification, keeping in mind NRC's role in these matters. However, we conclude that should a question arise in regard to NRC notification, the licensee should plan to err on the side of providing prompt notification.

We have concluded that the licensee 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 has reported that hydrogen gas generation is not a problem for Millstone-1. During normal operation, the reactor pressure vessel dome is filled with steam, which flows to the turbine. During reactor isolation, the dome may be automatically vented through the safety relief valves to the suppression pool. In addition, the reactor pressure vessel head has a vent line with a valve remotely operated from the control room.

The licensee response stated that the primary containment is nitrogen inerted per Technical Specification requirements and thus, hydrogen flammability is precluded.

We have concluded that the licensee response to IE Bulletin 79-08 item 10 is acceptable.

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

The licensee has reported in its letter dated May 14, 1979 that the Bulletin responses forwarded to NRC on April 24, 1979, and the various administrative and technical methods of implementing those responses have been carefully evaluated. The licensee concluded that no Technical Specifications changes are required at this time. We have concluded that the licensee response to IE Bulletin 78-08 item 11 is acceptable.

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 implications on Millstone-1 operations, and provide added assurance for the protection of the public health and safety during the operation of Millstone-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. NNECO letter, W. Counsil to B. Grier, dated April 24, 1979.

5. NNECO letter, W. Counsil to B. Grier, dated May 14, 1979.

6. NNECO letter, W. Counsil to B. Grier, dated June 15, 1979.