Nebraska Public Power District

COOPER NUCLEAR STATION
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August 7, 1979

Mr. Thomas A. Ippolito, Chief
Operating Reactors Branch No. 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: IE Bulletin 79-08, Request for Additional Information

Reference: Ippolito to Pilant letter dated July 20, 1979

Dear Sir:

This letter is written in response to the referenced letter requesting additional information concerning our response to IE Bulletin 79-08 dated April 25, 1979. The following items address the items as listed in the referenced letter.

Question

Item No. 1

1. Confirm that the review of item 1 of IEB 79-08 by all licensed operators and plant management and supervisors with operational responsibilities has been documented in your plant records.

Response

1. The review of item 1 of IE Bulletin 79-08 by all licensed operators and plant management and supervisors with operational responsibilities has been documented in plant records.

Question

Item No. 2

1. Your response indicates that you reviewed containment isolation of all valves whereas the Bulletin refers to all lines. Confirm that your review considered isolation of all lines penetrating containment.

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Response

1. The design of the BWR containment is such that all lines penetrating containment have valves which may be isolated as required. Our previous response stating our review of all containment isolation valves implied we reviewed isolation of all lines penetrating containment.

Question

Item No. 3

1. For the manual actions related to restart and continued RCIC operation, and for any other manual actions required, specify whether these actions are addressed by written procedures.

Response

1. The manual actions related to restart and continued RCI operation are addressed in written procedures.

Question

Item No. 4

1. Your response is incomplete. Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems.

2. Your response is incomplete. In addition, describe other instrumentation which the operator might have to determine changes in reactor coolant inventory, e.g., drywell high pressure, radioactivity levels, suppression pool high temperature, containment sump pump operation, etc.

3. Clarify your response to indicate whether operators have been instructed to utilize other available information to initiate safety systems. Provide your schedule for completion of this action.

Response

1. Reactor Water Level Instrumentation

   a. LIS-57 A&B (Yarway)
      Location: Rack 25-5
      Range: -150" to +60"
      (366.75" to 576.75" vessel height)
      Function: Trip RR Pumps, Close MSIV, and Closes Reactor Water Sample Valves ($\geq$-37")

   b. LIS-58 A&B (Yarway)
      Location: Rack 25-6
      Range and Function same as LIS-57 A&B

   c. LIS-72 A,B,C,D (Yarway)
      Location: Racks 25-5 & 25-6
      Range: -150" to +60"
      Function: Initiate RCIC & HPCI at ($\geq$-37")
      RHR, CS, DG & ADS ($\geq$-145.5")

   d. LIS-83 A&B (Yarway)
      Location: Racks 25-5 & 25-6
      Range: 0 to 60"
      (516.75 to 570.75" vessel height)
      Function: ADS Permissive ($\geq$12.5")

   e. LITS-59 A&B (Yarway)
      Location: Racks 25-5 & 25-6
      Range: -150" to +60"
      Function: Feeds LI-85 A&B on Panel 9-5, Control Room

   f. LITS-73 A&B (Yarway)
      Location: Racks 25-51 and 25-52
      Range: -100" to +200"
      (252.56" to 552.56" vessel height)
      Function: RHR 2/3 core height interlock ($\geq$-39")
      Feeds LI-91 A&B, Panel 9-3

   g. LT-70 (Gemac)
      Location: Racks 25-52
      Range: 0" to 100"
      (516.75" to 616.75" vessel height)
      Function: Feeds LR-98, Panel 9-5, Control Room

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h. LT-52 A,B,C (Gemac)
Location: Racks 25-5 & 25-6
Range: 0" to 60"
Function: Feedwater Control, Main TG & RFPT Hi level trips
Feeds LI-94 A,B,C on Control Room Panel 9-5
LR-97 on Control Room Panel 9-5

i. LT-61 (Gemac)
Location: Local Rack 25-5
Range: 0" to 400"
(516.75" to 916.75" vessel height)
Function: Feeds LI-86 on Control Room Panel 9-4

j. LIS-101 A,B,C,D (Gemac)
Location: Racks 25-5 & 25-6
Range: 0" to 60"
Function: Reactor Scram, Primary Containment Isolation at $\geq$12.5". Trips RCIC & HPCI at $\leq$58.5"

2. The following instrumentation could assist the operator to determine changes in reactor coolant inventory:

   a. Drywell equipment and floor drain sump flow recorders are provided in control room.

   b. Drywell equipment sump temperature indicator is provided in the control room.

   c. Mismatch between reactor feedwater flow and steam flow recorders and indicators.

   d. The suppression pool water level is detected by level switches, indicators and transmitters. Three suppression pool water indicators and one recorder are supplied in the control room on Panel 9-3. The recorder is a dual channel recorder, PC-LR-11 torus level, and PC-PR-512A drywell pressure. LI-10 provides torus indication over a range of -4.4 ft. to +6 ft. LI-12 provides narrow range indication over a range of -10 in. to +10 in. LI-13 provides narrow range indication over a range of -5 in. to +10 in. Should suppression chamber level reach a level of +5 inches two alarms will be annunciated. One from LT-12 "suppression chamber level high/low" which is activated at +2 in. (high) and -1 in. (low). The other from LT-10 "suppression chamber level high/low" which is activated at +5" (high) and -5" (low).

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e. Three primary containment and one wetwell pressure indications are provided in the control room.

f. Primary containment internal temperature is detected by 38 temperature elements of which four are used for wetwell pool temperature.

g. Drywell process radiation monitor which monitors particulate, gaseous, and iodine activities, plus provides the capability of a grab sample.

h. Main steam line high radiation and main steam line high flow alarms.

i. Reactor water cleanup high flow alarm.

j. High area temperatures (steam leak detection) alarms.

3. Operators have been instructed to utilize all available information to initiate safety systems. This is documented in plant training records.

Question

Item No. 6

1. It is not clear from your response that safety related valve positioning requirements were reviewed to ensure proper operation of engineered safety features. Please supplement your response to provide a commitment to conduct this review and a schedule for completion.

2. Please augment your response to indicate the extent to which position and locking device checks are performed for locked safety system valves.

3. Your response did not clearly indicate that all accessible safety related valves had been inspected to verify proper position. Nor was a schedule for performing the position verification for all safety related valves provided. Please supplement your response to provide this information.

Response

Item No. 6

1. Prior to our April 25, 1979 response to IEB 79-08, our safety related valve positioning requirements were reviewed to ensure proper operation of engineered safeguard feature systems. An I&E inspection team also independently inspected and performed a detailed review of the engineered safeguard feature systems on May 1-4 and 14-15, 1979. This inspection is described in IE Inspection Report 50-298/79-10. Two minor discrepancies were noted and have since been corrected.

2. The following excerpt is taken from our administrative procedure concerning valve seals:

   "The Valve Seals Log shall contain a list by system of the normally locked open/closed valves. These valves shall be sealed in such a manner as to prevent their being operated without destroying the seal. The seal will consist of a lead-wire, numerically numbered seal.

   The Shift Supervisor will maintain the Valve Seals Log. Whenever a seal is broken for any reason, the Shift Supervisor shall be notified and the seal returned to him. He will then cross out the seal number as listed in the log and record the valve with the missing seal until the seal is replaced in the Shift Supervisor's log as a red arrow item.

   Upon return of the valve to its normal locked open or closed position, the Shift Supervisor shall be notified. He will then issue a new seal with the next higher number for that particular system and list it in the Valve Seal Log. Each system is issued a block of seals in numerical order."

   The valve seals broken during surveillance testing or maintenance are replaced and a specific sign off is required. All seals are checked as part of the valve lineups performed after an extended outage or at the discretion of the Operations Supervisor.

3. Following our April 1979 outage which was in progress at the time of our initial response to IEB 79-08, all safety related valve positions were verified to be in the proper position. As described above, this position verification was independently checked by an I&E inspection team.

Question

Item No. 7

1. In your discussion regarding systems designed to transfer radioactive gases and liquids, no explicit discussion is presented regarding valve action or resetting safety features instrumentation. Provided assurance that inadvertent transfer of radioactive gases or liquids out of containment will not occur or resetting safety feature instrumentation.

Response

1. At the present time our procedures caution the operator to ensure that the problem that caused the containment isolation has been rectified, prior to resetting the instrumentation. We will review our procedures and update them as necessary to provide additional guidance concerning parameters which should be checked prior to resetting isolation instrumentation. This review and update will be completed by September 15, 1979.

Question

Item No. 8

1. We understand from your response that operability is verified for redundant safety related systems prior to removal of any safety related system from service. Since you may be relying on prior operability verification within the current technical specification surveillance interval, operability should 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. Please supplement your response to provide a commitment that you will revise your maintenance and test procedures to adopt this position.

2. It is not clear from your response that all involved reactor operational personnel in the oncoming shift are explicitly notified about the status of systems removed from or returned to service. Please indicate how this information is transferred at shift turnover.

Response

1. The Technical Specifications at Cooper Nuclear Station require that redundant systems be tested immediately and periodically thereafter when a system is made or found inoperable. When a system is to be made inoperable for maintenance, station procedures require that the redundant systems be verified operable with an actual system test prior to removal of that system. This generally involves a visual inspection of the equipment being tested, however the control room operator doing the testing has enough information immediately available to him to determine system operability.

2. The following items are required by our procedures as a minimum turnover for all licensed operators:

   a. Review status of plant and operating procedures in progress.

   b. Review safety system status panel.

   c. Review respective log back to their last working shift.

   d. Review surveillance tests in progress.

   e. Significant changes in routine operation which have occurred during the last two shifts.

   f. Review any abnormal circumstances.

   The shift supervisor must also review all maintenance work orders in progress, review any new special orders issued since the last working shift, and review the Night Orders Log.

   Events such as removing from or returning to service of systems are typical entries required by procedures to be made in the Control Room and Shift Supervisor's Logs. In addition, the safety system status panel is provided on the reactor operator's panel. This panel is checked at each shift turnover and it indicates whether an engineered safety feature system or loop of that system is in service, in test, or out of service. Our test procedures have specific steps in them through which the status of the panel is updated.

Question

Item No. 9

1. Amend your response to provide assurance that your procedures stipulate that NRC will be notified any time the reactor is not in a controlled or expected condition of operation.

Response

1. Our station procedures stipulate that NRC will be notified any time the reactor is not in a controlled or expected condition of operation.

Sincerely,

J. M. Pilant
Director of Licensing
and Quality Assurance

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