Central files

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office

CEDAR RAPIDS, IOWA

April 27, 1979

JAMES A. WALLACE
VICE PRESIDENT – GENERATION

50-331

Mr. James G. Keppler, Director
Office of Inspection and Enforcement
Region III
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Re: Duane Arnold Energy Center

Subject: Response to IE Bulletin 79-08

File: A-101a

Dear Mr. Keppler:

In response to IE Bulletin 79-08 dated April 14, 1979, attached are our responses to Items 1 through 10. Our response to Item 11 will be sent to you by May 18, 1979 as required by the bulletin.

These responses are the result of the review that has been conducted over the past 10 days. This review was based on the information available in IE Bulletins 79-05 and 79-05A regarding the TMI-2 incident and the existing procedures and documents including FSAR safety analysis at the Duane Arnold Energy Center.

We anticipate that as a result of the training sessions which will be conducted as described in our response to Item 1, additional procedural revisions may be required in order to assist plant personnel in their operational responsibilities. We will remain in contact with you regarding any significant items that may arise.

Very truly yours,

J. A. Wallace

J. A. Wallace
Vice President-Generation

JAW/DLM/n

cc: U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Division of Reactor Operations Inspection
Washington, D. C. 20555

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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.

The description of circumstances detailed in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI incident included in Enclosure 1 to IE Bulletin 79-05A have been discussed and analyzed in an informal manner with all personnel at DAEC with operational responsibility. In order to formalize and document discussions of this situation, the following steps are being taken:

1. Copies of IE Bulletins 79-05 and 79-05A together with appropriate PWR system sketches and the DAEC response to this IE Bulletin have been placed in a binder in the Control Room. All licensed operators and plant management and supervisors with operational responsibility shall read this material and indicate on an accompanying sign-off sheet that they have done so. This step shall be completed prior to May 14, 1979.

1. (Cont.)

2. A series of training sessions will be held for all licensed operators and plant management and supervisors with operational responsibility to thoroughly familiarize them with the TMI incident and its potential applicability to DAEC. Particular emphasis shall be placed on the specific concerns expressed in Action Item 1 above. Participation in these sessions shall be documented as per the requirements of Section 5.0 of the DAEC Training Programs Administrative Manual, "License Training Program". This step shall be completed prior to July 1, 1979.

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.

Containment isolation occurs upon reactor vessel low water level or high drywell pressure signals prior to or simultaneous with initiation of emergency core cooling and safety injection systems. The isolation includes all lines whose isolation does not degrade needed safety features or cooling capability. Isolation valves will remain closed, even when the initiating signal clears, at which time the containment isolation system can be manually reset by operator action. The training sessions described in the response to Item 1 above will specifically address this point.

Plant operating procedures require that upon containment isolation the operator verifies that all required automatic isolation has occurred.

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.

In the event of loss of feedwater with the reactor at power, systems available for heat removal in addition to CRD system flow to the vessel which continues throughout the transient, include the following:

High Pressure Coolant Injection (HPCI)

Reactor Core Isolation Cooling (RCIC)

Automatic Depressurization (ADS)

Low Pressure Coolant Injection (LPCI)

Core Spray (CS)

No operator action, other than verification of proper actuation and functioning of the above systems is required, although operator action can be taken to control reactor water level and reactor coolant system pressure to minimize cycling of the associated systems as trip and initiation set points are reached.

The basic sequence of automatic actuation of the above systems is as follows under the assumed loss of feedwater transient.

Loss of feedwater

Vessel water level decrease to +12" Barton (lo level)

Reactor Scram

Vessel water level decrease to -38.5" Yarway (lo-lo level)

Recirc Pump Trip

MSIV closure

HPCI start

RCIC start

LPCI loop select logic actuated

3. (Cont.)

Both HPCI and RCIC take suction from the Condensate Storage Tank (CST) and discharge to the reactor vessel through the Main Feedwater System piping downstream of the motor operated Feedwater System stop check valves. In the event that CST level decreases to 10,000 gallons or Suppression Pool level increases to 5" above normal, HPCI suction will automatically be transferred to the Suppression Pool. RCIC suction must be manually transferred via remote manual control switches in the Control Room to the Suppression Pool as detailed in DAEC Operating Instruction 50, RCIC System.

In the event that reactor water level continues to decrease and reaches -139.5" Yarway, both the LPCI and CS systems start. At 450 psig reactor coolant system pressure, injection valves for each of the systems open, establishing flow to the vessel. LPCI injection is via either of the reactor coolant system loops. The B loop is preferentially selected unless loop select logic senses a break in that loop and automatically repositions viaving to the A loop. The Core Spray System injects water directly into the vessel via the core spray spargers.

Drywell high pressure also initiates HPCI, LPCI, and CS. The systems function as previously described except that upon HPCI turbine trip at high reactor vessel water level, restart will not occur until reaching the low-low reactor vessel level setpoint.

Manual operator action directly related to loss of the main feedwater system that can be taken includes:

Remote manual flow control of both the HPCI & RCIC systems

Actuation of the safety relief valves to control system pressure

3. (Cont.)

flow control to maintain reactor water level can be achieved by adjusting the setpoint of the Flow Indicating Controller for the desired system. Reactor vessel pressure can be controlled utilizing ADS valves by actuating the respective control switch in the Control Room to the open and close position.

4. Describe all uses and types of vessel level indication for both automatic and manual initiation of safety system. 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 system.

Instrumentation at DAEC which senses water level in the reactor vessel is employed for the follwoing purposes:

1. To provide inputs to the Feedwater Control System
2. To provide inputs to the Reactor Protection System
3. To provide inputs for initiation of the Emergency Core Cooling Systems
4. To provide Control Room indication of vessel water level during all modes of reactor operation.

Reactor vessel water level monitored by 7 indicators and two 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.

These water level measurement devices have operated in BWR plants for 20 years. Tests of BWR water level instrumentation under simulated steam and water line breaks have been conducted showing satisfactory performance.

The range of reactor vessel water level from below the top 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/or 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.

4. (Cont.)

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

Reactor Core Isolation Coolant System (RCIC)

High Pressure Coolant Injection System (HPCI)

Core Spray System (CS)

Residual Heat Removal/Low Pressure Coolant Injection (RHR/LPCI)

Automatic Depressurization System (ADS)

Primary Containment Isolation System (PCIS)

All systems automatically initiate on low reactor water level. In addition, the RCIC and HPCI systems shutdown on high reactor vessel water level. In all cases, these systems automatically restart if low reactor level is again reached.

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 (SRV) Discharge High Temperature

Safety Relief Valve (SRV) Actuation Alarm

High Feedwater Flow Rates

High Main Steam Flow

High Containment and Equipment Area Temperatures

High Differential Flow-Reactor Water Cleanup System

Abnormal Reactor Pressure

High Suppression Pool Water Level

High Drywell and Containment Sump Fill and Pumpout Rate

Valve Stem Leakoff High Temperature

4. (Cont.)

An example of the use of this additional information by the operator is as follows: Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High SRV discharge temperature would pinpoint loss of coolant via an open valve.

Other instrumentation that can signal abnormal plant status but not necessarily 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 (Amperes) to Pump Motors

Operators will be instructed to utilize other available information when initiating safety systems. These instructions will be given during the training sessions described in the response to Question 1 above.

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.

A review of the DAEC Plant Emergency Instructions indicates that some precautionary instructions to the operators about overriding automatic actions of engineered safety features are included. However, not all situations or systems are included. These procedures will be revised, as required, to include precautionary instructions on overriding automatic actions for all ECCS systems. In addition, the training sessions as scheduled in the response to Question 1, will include instructions in this area.

The DAEC Emergency Procedures provide the operator with indications of other plant parameters for evaluating plant conditions. The procedures will be reviewed and revised as necessary to provide the operator with additional information and instructions regarding singular reliance upon vessel level indication alone for manual actions, but also to examine the other plant parameters in evaluating plant conditions. This additional information and instruction will be provided in training sessions as scheduled in the response to Question 1.

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.

Valve positions for normal system operation for all safety related valves are indicated on a "Prestartup Valve Checklist" which is attached as an appendix to each System Operating Instruction in the DAEC Operations Manual and is utilized at each startup for the individual system. Valve checklists are utilized by an operator to verify valve position. These checklists are then reviewed and approved by the Shift Supervising Engineer prior to system startup

All safety related valve positions are presently being reviewed by completing the individual valve checklists to the extent possible due to plant accessibility requirements. Where valve are inaccessible and no remote position indication is available shall be made by reviewing the latest checklists whick were completed in February, 1979. This review will be completed prior to May 14, 1979. As plant conditions permit, the inaccessible valve positions will be verified.

In February, 1979, prior to startup following DAEC's extended outage, a detailed review was completed of the Prestartup Valve Checklist for all safety related systems. This review included examination of system P&ID's, an extensive walk-down of each system and system operational experience. As a result of this review, it has been determined that the positioning requirements of safety related system valves as reflected on the Prestartup Valve Checklists reflect the conditions required for proper operation of these systems.

6. (Cont.)

In the event that any valve position changes are effected either due to Surveillance Test Procedures (STP's) or maintenance activities, documentation of return to normal operating position is verified. In the case of an STP, the Shift Supervising Engineer verifies that the system has been returned to its normal operating condition prior to signing the STP. In the case of maintenance activities, a Maintenance Action Request (MAR) requires that appropriate testing be performed at completion of any maintenance to insure that the system has been returned to normal operating condition. An Inspection and Test Report is attached to all Safety Related MAR's and the MAR is signed off by the Shift Supervising Engineer. These requirements are documented in DAEC ACP's 1401.4, "Control of Plant Work", and 1408.3, "Surveillance Program".

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.

A review of all procedures for systems designed to transfer potentially radioactive gases and liquids out of the primary containment was performed to assure that inadvertent radioactive releases do not occur.

The following systems are designed to transfer potentially radioactive gases and liquids out of the primary containment:

1. Drywell equipment sump
2. Drywell floor drain sump
3. Main steam
4. Reactor water cleanup
5. Containment atmospheric control
6. Reactor water sample
7. Residual heat removal
8. Main steam isolation valve leakage control

The main steam system, including main steam line drain valves, and the reactor water sample lines isolate on high radiation in the main steam lines. The containment atmospheric control system isolates on high radiation in the reactor building. These are the only systems with high radiation signal interlocks. However, all the system listed above, except the main steam

7. (Cont.)

isolation valve leakage control system isolate on a containment isolation signal prior to or simultaneous with initiation of emergency core cooling. The main steam isolation valve leakage control system is a manually operated, normally isolated system with no initiating automatic function.

Continued operability of the primary containment isolation system is assured by frequent calibration and functional testing of initiating instrumentation. Additionally, the operable isolation valves that are power operated and automatically initiated are tested for simulated automatic initiation and closure times. All testing is performed in accordance with the DAEC Technical Specifications.

Instrumentation that initiates primary containment isolation will only reset when the initiating signal clears, but the primary containment will remain isolated until the containment isolation system is manually reset by operator action. Procedures will be developed which will prevent the operator from manually resetting the containment isolation system until it is verified that radioactive gases and liquids will not be inadvertently released.

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 for 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.

a. Upon declaration that a safety-related system is inoperable, or prior to the removal of any safety-related system from service, the operability of redundant safety-related system(s) is verified. This is done by means of "Inoperable Packages" which exist for each system in the Control Room under the cognizance of the Shift Supervising Engineer. These packages detail the increased surveillance and/or operability requirements per the DAEC Technical Specification whenever a safety-related system has been declared inoperable. The appropriate package is consulted and the required testing or inspection initiated. The declaration of the system inoperability is recorded in the Shift Supervising Engineer's Log and the Operating Log as described in DAEC ACP 1404.4, "Operating Logs".

b. Verification of operability of all safety-related systems as they are returned to service following maintenance or testing is as described in the response to item 6 above.

c. Whenever a safety-related system is to be removed from or returned to service, this fact is reported to and approved by the Shift Supervising Engineer (by means of an STP or MAR in accordance with the ACP) discussed in the response to items 6 above. This fact is then recorded in the Shift Supervising Engineer's Log and the Operating Log in accordance with ACP 1404.4.

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.

DAEC Administrative Control Procedures (ACP) 1401.6 and 1401.7 and Quality Assurance Manual Directive 1316.2 address existing Technical Specification reporting requirements and specify that the Shift Supervising Engineer as well as other cognizant plant supervisors are responsible for prompt notification of designated management personnel and the NRC in the event of a reportable occurrence. In addition, the Preparedness Plan Implementation Procedures provide guidance regarding notification of off-site agencies including the NRC based upon an evaluation as to the seriousness of the condition. Further clarifying instructions will be issued providing guidance to the Shift Supervising Engineers and DAEC management personnel with respect to prompt NRC notification (i.e., within 1 hour) of conditions when difficulty is being experienced in placing the reactor in a stable condition.

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.

During normal operation at DAEC, the reactor pressure vessel dome is filled with steam which flows to the turbine. During reactor isolation, the dome is automatically vented through the SRV's to the suppression pool as necessary. In addition, the reactor pressure vessel head has a vent line with a valve remotely operated from the Control Room which vents to the drywell equipment sump.

Operating procedures require that hydrogen concentrations be maintained below 4% during post LOCA conditions. This is performed using the containment atmosphere dilution system which adds nitrogen to the primary containment as required.