CP&L

Carolina Power & Light Company

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Mr. James P. O'Reilly, Director
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
RESPONSE TO IE BULLETIN 79-06A

Dear Mr. O'Reilly:

Attached you will find CP&L's response to Items 1 - 12 of Bulletin 79-06A concerning operational errors and system misalignments identified during the Three Mile Island incident.

I trust that this information is suitable for your use.

Yours very truly,

DBWaters for BJ Furr

B. J. Furr
Manager
Generation Department

CSB:jmb*

Attachment

cc: Nuclear Regulatory Commission
Office of Inspection & Enforcement

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CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
DOCKET 50-261
RESPONSE TO IE BULLETIN 79-06A

Item 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 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 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 shall participate in this review and such participation shall be documented in plant records.

CP&L RESPONSE

Prior to receipt of IE Bulletin 79-06 and 79-06A, a comprehensive review of the details (as many as were available) of the TMI incident had already started. Specifically, on Friday, March 30, the personnel at the CP&L simulator had been contacted and asked if this incident, given what we understood of the incident, could be duplicated for Robinson. However, because of basic design differences and a lack of event details, total duplication of this event was not possible. On Monday, April 2, having gathered substantially more information concerning the TMI event, a meeting was held by the Plant Manager to discuss the event with all plant management personnel available in the morning. At their suggestion, a similar meeting to discuss the event was held by the Plant Manager for all employees who were available at 1500 hours that afternoon. Approximately 60% of all plant employees attended this meeting. In the days that followed, a corporate level review team was established and a plant representative was assigned to participate on this review team in the overall review of the TMI incident. Although several specific items have been reviewed

CP&L RESPONSE - Continued

to date, three major discussions which have transpired recently were: (1) A meeting at the General Office to review the TMI incident and highlights of the Plant Emergency Plan with corporate management on April 13; (2) A meeting to discuss the operational details of the TMI incident with participants from the Plant, CP&L General Office and Westinghouse in Pittsburgh, Pennsylvania on April 18 and 19; (3) a review of the TMI incident by NRC Region II Inspection and Enforcement personnel with plant operations personnel on April 23.

All licensed operators and plant management and supervisors with operational responsibilities have participated in these reviews. The participation of these personnel will be documented in plant records. As a result of these reviews, the problems of overriding of automatic actuation of engineered safety features and basing of operational decisions on a single plant parameter indication have been highlighted. Detailed procedure changes will be developed as discussed in subsequent responses below to ensure these items are adequately covered.

Item 2. Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

b. Operation action required to prevent the formation of such voids.

c. Operator action required to enhance core cooling in the event such voids are formed (e.g., remote venting).

CP&L RESPONSE

2a. The primary indication of void formation in the primary coolant system is provided when the pressurizer pressure falls below the hot leg saturation pressure. However, during a loss of coolant event, void formation in the primary coolant system would be expected with two exceptions: i) the loss of coolant is being caused by a stuck open pressurizer relief valve which closes or is isolated before the system depressurizes to hot leg saturation, or, ii) the reactor coolant system reaches an equilibrium pressure above hot leg saturation, when the safety injection flow equals the break flow. Thus, in these two specific cases, confirmation of no voids in the system will be apparent by the pressure in the pressurizer. In the remaining cases, the engineered safeguards system has been designed to cope with voiding. Thus, it is not necessary to be able to recognize void formation in those cases.

Instructions and discussion relating to recognition of the possibility of forming voids in the primary coolant system points out that maximum charging flow and reactor coolant pump seal injection flow cannot maintain the pressurizer water level under certain LOCA conditions. Further, the instructions point out that the pressurizer will be completely drained following a double ended tube rupture.

2b. These are discussed under our Emergency Operating Instructions. Immediate actions, prior to diagnosis of the specific accident classification, which tend to prevent formation of voids, include:

1. Verification that reactor trip and safety injection have occurred.

2. Verification that feedwater is being supplied to the steam generators.

For some LOCA cases, no operator action will prevent the formation of voids in the primary coolant system. The emergency safeguards system was designed to recover and cool the core following various degrees of primary coolant system voiding, depending on the break size and location.

CP&L RESPONSE - Continued

Procedures are being reviewed to include the following:

1. Prevent steam void formation by checking reactor coolant pressure and temperature following reactor trip to ensure that the reactor coolant temperature in the hot and cold legs is at least $50^{o}F$ below the saturation temperature.

2. Provide for termination of operation of HPI Systems only when conditions described in the response to Item 7 are met.

   Final implementation of revisions to procedures will be accomplished prior to unit startup at the end of the current refueling outage.

2c. Referring to actions under Items 2a and 2b, our Emergency Operation Instructions adequately describe the necessary operator action to enhance core cooling of the primary coolant system that has been voided due to either a loss of coolant, loss of secondary coolant, or steam generator tube leakage accident. However, we will continue our review of this item to determine whether additional actions are necessary.

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Item 3. For your facilities that use 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 reaches the low setpoint, safety injection would be initiated regardless of the pressurizer level. In addition, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation setpoint whether or not the level indication has dropped to the actuation setpoint.

CP&L Response

Westinghouse recommended to H. B. Robinson Plant, by telephone on April 7, 1979 and by letter, April 10, 1979, that our operators monitor pressurizer pressure along with other important variables. In addition, Westinghouse recommended that safety injection be manually initiated when the pressurizer pressure drops below the initiation setpoint.

In accordance with the requirement of Item 3, the low pressurizer level bistables have been tripped. Consideration is being given to a permanent plant modification to remove the pressurizer level signal as a safety injection actuation requirement and converting the low pressurizer pressure input to the safety injection initiation to a two out of three logic. This will provide sufficient protection in accordance with designs on later Westinghouse plants, avoid spurious actuation of safety injection, and provide more positive means of performing periodic testing of the trip channels.

Item 4. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to permit 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.

CP&L Response

The containment isolation consists of Phase A and Phase B isolation. Phase A isolate all non-essential process lines (except the N_2 supply to the pressurizer relief tank and reactor coolant drain tank) but does not affect safety injection, containment spray, component cooling, steam and feedwater lines. The procedures to isolate the N_2 supply to the PRT and RCDT are being revised to provide for manual isolation of these lines. Phase A isolation is initiated by (automatic or manual) safety injection initiation and also can be initiated manually. Phase B isolates all remaining process lines (except safety injection, containment spray lines and auxiliary feedwater) and is initiated by a 2/4 Hi-Hi containment pressure or manual containment spray actuation.

The design criteria for containment isolation is such that it is not reset by the elimination or resetting of the initiating signals, for example by resetting safety injection. Containment Isolation can only be reset by manual controls on the main control board.

Control features are provided for the containment isolation valves such that:

a) The valves will remain in the closed position if the trip signal is reset.

b) The containment isolation signals override all other automatic control signals.

c) Each valve can be opened or closed normally after the containment isolation signals are reset

Thus, the design criteria provides for containment isolation and includes provisions to provide that containment isolation is not degraded by reset of initiating signals.

Item 5. For facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require 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 can be limited by such action.

CP&L Response

Our Auxiliary Feed Water System is automatically initiated as a result of the following conditions:

a. Safety Injection initiation.

b. Loss of offsite power.

c. Loss of two main feedwater pumps.

d. 2/3 level channels sensing low-low level (15%) on 1/3 steam generator will automatically start both motor driven AFW pumps and open discharge valves.

In addition, 2/3 level channels sensing low-low level (15%) on 2/3 steam generators will automatically start the steam driven AFW pump and open discharge valves. Thus, this item does not apply to H. B. Robinson Unit No. 2.

Item 6. For your facilities, prepare and implement immediately procedures which:

a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and

b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to below the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) remain stuck open.

CP&L Response:

6a. Pressurizer Relief Valve Open Indicators

1. H. B. Robinson 2 has 2 power operated relief valves. These 3 inch valves are each isolated with a normally open 3 inch isolation valve. Each of these 4 valves have control modules on the main control board which provide manual (open/close) control and position light indication or open/close position via limit switches.

2. The lines with each pair of valves come together in a common header where discharge temperature is measured. Temperature is displayed on the Main Control Board and an alarm sounds on high temperature (20°F above normal ambient). This 6" line is headered together with lines from the 3 safety valves (which have similar indication and alarm on discharge temperature).

3. This 12" header passes fluid to the PRT which is instrumented for temperature, pressure and fluid level. All instrument channels give main control board indication and alarm and are as follows:

Liquid Temp.: Alarm at 150°F

Pressure: Alarm at 5 psig

Level: Alarm high 83%

low 68%

In addition to these three indications of actual discharge, the pressurizer pressure channels which cause the valves to open automatically will also turn on main control board alarms on high pressure.

Thus, several sources of diverse indication exist to assist the operators in determining if a pressurizer PORV is open.

6b. Plant procedure EI-1 (Incident Involving Reactor Coolant System Depressurization) will be revised to reflect requirement of item 6b prior to plant start at the end of current refueling outage.

Item 7. 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. For example, if continued operation of engineered safety features would threaten reactor vessel integrity then the HPI should be secured (as noted in b(2) below).

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of 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 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 F and the length of time HPI is 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 (RCP) 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 parameters indications in evaluating plant conditions, e.g., water, inventory in the reactor primary system.

CPL Response:

7a. An indepth review of appropriate operating procedures and training instructions has been completed.

Criteria and guidance have been included in our procedures for termination of safety injection following secondary side breaks which lead to a primary system cooldown. In this case continued operation of safety injection could lead to conditions which potentially could exceed reactor vessel pressure criteria. Similarly, following a steam generator tube rupture, criteria and guidance have been prepared for termination of safety injection to reduce the quantity of primary coolant which passes to the secondary side of the steam generator.

In addition to the above, we are reviewing our procedures to assure the following:

1. Delay reset SI action until just prior to the first operator action which will change the status of the ECCS equipment automatically actuated by the S signal. The SI reset action should assure that containment isolation is maintained.

2. Following an "S" signal, if offsite power is available, the operator will determine that the diesels are immediately available and operable.

3. For all cases where the RCS pressure equilibrates above the LHSI pump shut-off head, the LHSI pump is to be stopped and placed in the stand-by mode.

4. In all procedures where the plant is controlled by normal makeup and letdown and RCS pressure cannot be maintained above the set-point for SI actuation, then procedures should provide for the following:

   A. Manually initiate Safety Injection.

   B. Go to plant emergency operator instructions.

5. In those cases where plant recovery and depressurization steps are conducted using Emergency Instructions, procedures will be reviewed with respect to manual or automatic steps which were placed there as a result of RCS cold overpressurization (Appendix G) concerns. Cold overpressurization protection systems should not be activated for those cases. The operation of these systems could lead to an inadvertant RCS depressurization process through operation of the Pressurizer PORV's.

6. Emergency operation instructions will be reviewed to assure that the operator monitors wide range RCS temperature and pressure, steam pressure, steam generator water level, containment pressure, RWST level, condensate storage tank level, pressurizer level and boric acid storage tank level. Verification of indicator operation will also be made.

7b. The above recommended actions will be reviewed for incorporation into appropriate plant procedures prior to plant startup at the end of the current refueling outage.

7c. H. B. Robinson #2 is a 3-loop plant. Our current procedures do not call for Reactor Coolant Pumps (RCP) tripping in the event of HPI initiation with the RCPs operating. However, situations where one or more RCPs may need to be shut down will be reviewed to ensure that procedures adequately address the concerns associated with long term cooling requirements versus requirements for forced circulation.

7d. Modification to Emergency Operator Instructions are now being made to ensure operator action based on several parameter indications. As a minimum, the operator is provided with indications of:

Wide Range RCS Temperature and Pressure
Steam Pressure
Steam Generator Water Level
Containment Pressure
RWST Water Level
Condensate Storage Tank Level
Pressurizer Water Level
Boric Acid Storage Tank Level

Item 8. 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 or 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.

CPL Response:

We have reviewed this item, and believe that it is adequately addressed by existing procedures and operational methods as follows:

a. Safety related valve positions are verified and documented by use of System Line-ups.

b. Operational work permits, periodic tests and operating procedures insure that safety related valves are returned to their correct positions following manipulation due to maintenance, testing or plant start-up.

c. Shift Foreman's Minimum Equipment List (completed each 8-hour shift) provides surveillance of operable safety systems.

Item 9. 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 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.

CP&L Response

The operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of containment have been reviewed to assure that undesired pumping, venting, or other releases of radioactive liquids and gases will not occur inadvertently. Our present design allows pressure relief valves and vacuum relief valves to open automatically on reset of ventilation isolation. Modifications are being implemented during our present refueling outage to assure continued automatic isolation on resetting of ventilation isolation. No transfer systems start automatically by the resetting of engineered safety features.

Item 10. Review and modify as necessary your maintenance and test procedure 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.

CP&L RESPONSE

The review required in Paragraph 10.a through c is in progress and will be completed and any identified changes required by the review will be implemented by the end of the present refueling outage.

Item 11. 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.

CP&L RESPONSE

The applicable procedure will be revised consistent with Item 11 prior to startup following the current refueling outage.

Item 12. 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.

CPL Response:

The engineered safeguards are designed and analyzed to meet the limits of 10 CFR 50.46 which require that the hydrogen generation from clad water reaction in a LOCA be limited to less than 1% of the clad metal, and nowhere exceed 17% of the clad thickness.

The modes for removing hydrogen from the reactor coolant system are:

1. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if the reactor coolant pump is operating.

2. Hydrogen in the pressurizer vapor space can be vented by power operated relief valves to the pressurizer relief tank.

3. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system. Our waste gas system has 4 tanks storage capacity of 525 SCF each.

4. In the event of a LOCA, hydrogen would vent with the steam to the containment.

If for some reason a non-condensible gas bubble becomes situated somewhere in the primary coolant systems, there are many options for continued core cooling and removing the bubble.

With a gas bubble located in the upper head several methods of core cooling are unaffected. The steam generator can be used to remove decay heat using reactor coolant pump forced flow or natural circulation. The safety injection system can be used to cool the core while venting through the pressurizer power operated relief valve. Core cooling by any of these methods can proceed indefinitely if the primary coolant pressure is held constant. If a lower system pressure is desired, a controlled depressurization will allow the bubble to grow slowly until it uncovers the top of the hot legs.

This controlled depressurization can be performed in two ways:

1. If the reactor coolant pumps can be operated, depressurization can be performed with a steam bubble, using pressurizer power operated relief valve. Extra control can be achieved with the pressurizer heaters and sprays. As the bubble grows to the top of the hot leg, small bubbles are carried through the system. Degassing is done with the spray line and/or the Chemical and Volume Control System. The steam generators will carry away decay heat.

2. If the reactor coolant pumps cannot be operated or their operation is undesirable (as per item 7.c) the pressurizer can be made water solid with the safety injection pumps running and the power operated relief valve and/or vent valve open. Depressurization is controlled by judicious use of the various valves, lines and pumps available in the safety injection system and by adjusting the pressurizer relief valve and/or vent valve. As the bubble grows to the top of the hot leg, it slides across the hot leg and up into the steam generators. As depressurization continues the gas bubbles grow in the steam generators and upper head but the core remains covered and cooled by safety injection water. If there is enough gas, the pressurizer surge line would eventually be "uncovered". Some of the gas would burp into the pressurizer and out of the valve. This burping process would continue until the system were at the desired pressure. At that time the current cooling mode could be continued or the system could be placed in an RHR mode.

Note that a gas bubble cannot be located in the steam generator with the reactor coolant pumps running. If a gas bubble forms in the steam generator during natural circulation, the reactor coolant pumps could be turned back for degassing or safety injection flow could be initiated with the power operated relief valve open.

We should also note that the gas bubbles cannot uncover the core in the above depressurization schemes because it will always tend to float to the top of the system and it cannot compress water.

We have reviewed plant operating instructions to assure adequate post-accident containment sampling circulation, and means of hydrogen removal which may be released to the containment during a transient or accident condition.

Item 13. Propose changes, as required, to those technical specifications which must be modified as a result of your implementing the above items.

CPL Response:

Any revisions to Technical Specifications required as a result of the reviews conducted in Items 1-12 will be submitted by the end of the current refueling outage.